行政院所屬各機關因公出國人員出國報告書

(出國類別：實習)

赴比利時SCK-CEN研究所實習核設施除役技術

服務機關：行政院原子能委員會核能研究所

出 國 人：職　　稱：薦任助理研究員

　　　　　姓　　名：郭鴻達

出國地區：比利時

出國期間：91年11月2日至91年12月7日

報告日期：92年2月6日

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Dismantling of the BR3 PWR Auxiliary and Primary Loops

- Overview of Recycling Technologies for Decommissioned Materials: Lessons Learned during the Dismantling of a Reactor
- Nuclear and Non-Nuclear Safety Aspects in Nuclear Facilities Dismantling: The Example of a PWR Pilot Decommissioning Project
- Dismantling of the BR3 Reactor Pressure Vessel

5. 比利時拆除混凝土 free release 標準：1 份
6. 除役受訓課程講義：1 份

摘　要

此次奉派前往比利時 SCK-CEN 研究所 BR3除役計畫學習核設施除役相關技術，學習期間自 91 年 11 月 2 日起至 91 年 12 月 7 日止，為期 36 天。在 SCK-CEN 學習期間規劃進行的工作項目有以下諸項：

1. 了解 BR3 計畫各期之執行狀況與其工作成果
2. 參與 BR3 計畫的部分工作並交換工作經驗
3. 蒐集除役規劃相關資訊
4. 蒐集歐洲核能工業商情
5. 現場工作見習
6. 參加 BR3 計畫舉辦的除役規劃訓練課程

在 SCK-CEN 學習期間完全依照 BR3 計畫之工作模式，並與 BR3 計畫人員一同工作。在計畫主持人 Vincent Massaut 先生完成了計畫執行狀況之簡報後，立即被邀請投入工作成為工作團隊組織成員。學習期間除了實際參與其規劃設計工作外（乏燃料運送程序審查），亦一同進入管制區見習 BR3 乏燃料裝罐作業，與運送至 Belgoprocess 儲存場之操作情形；並受邀參加其工作檢討會。

學習期間所得成果如下：

1. 充分了解 BR3 除役計畫的工作成果並吸收其工作經驗
2. 了解 BR3 工作團隊之分工與計畫管理模式
3. 蒐集 BR3 工作計畫書與相關除役、拆除與除污等相關資訊（約 1000 頁）
4. 與歐洲核能從業人員進行面對面的工作經驗交換並聽取其建議，有利本所日後老舊設施除役規劃。

一、目的

此次奉派前往比利時 SCK-CEN 研究所之 BR3 除役計畫學習核設施除役相關技術。學習期程自 91 年 11 月 2 日至 91 年 12 月 7 日，為期 36 天。

此次前往 SCK-CEN 研究所實習的目的有以下三項：

1. 在除役工作之規劃與執行方面，希望藉由現場觀摩與人員訪談了解 BR3 除役計畫的工作規劃與執行經過。
2. 在除役與除污的工作經驗方面能與歐洲核能先進國家之從業人員有相互交流之機會，以期能吸收其經驗或建議對本所用過燃料池清理或其他核設施除役等相關工作有所助益。
3. 歐洲核能相關工業之商情蒐集

在 SCK-CEN 學習期間所進行的工作項目有以下諸項：

1. 了解 BR3 計畫各期之執行狀況與其工作成果
2. 參與 BR3 計畫的部分工作並交換工作經驗
3. 蒐集除役規劃相關資訊
4. 蒐集歐洲核能工業商情
5. 現場工作見習
6. 參加 BR3 計畫舉辦的除役規劃訓練課程

二、過程

出發前完成之準備工作有：

①從 SCK-CEN 之網站上蒐集 BR3 拆除計畫之相關資料，對 BR3 核電廠之系統與拆除計畫執行至今之狀況有初步之了解（圖一）。

②組長李定一先生提示應加強資料蒐集與經驗交換之重點工作。工作項目包括：

- TRR 燃料池廢料處理相關技術
- 破損燃料之處理技術
- 燃料池整建
- 水泥塊廢料之除污技術與排放標準等方面之技術
- 商情資料蒐集與經驗交換。

③ 彙總本所以往在水下切割、輻射屏蔽與水下監控等與拆除作業或水下操作之困難，並將之轉譯成英文並佐以相關圖說、相片與影片。以期能在最短之時間內讓歐洲的核能同業們充分了解我等之技術能力與目前之瓶頸，並期望能提出最適當之建議。

④ 自費準備禮物共十份（古畫郵票冊 2 份，粽子香包飾品 8 份）分送重要計畫幹部，以促進文化外交。

在 SCK-CEN 實習期間：

在到達 SCK-CEN 後，由計畫主持人 Vincent Massaut 先生作 BR3 計畫之執行狀況簡報，並介紹認識 BR3 各相關工作之負責人員。在實習期間完全依照 BR3 計畫日常工作之模式（08：00 AM~17：00 PM)。並立即被安排成為 BR3 工作團隊之一員。

實習期間除了實際參與其規劃設計工作外，亦一同進入管制區見習 BR3 乏燃料裝罐運至 Belgoprocess 儲存場之

操作情形並於工作檢討會中被邀提出意見參與檢討工作。

總括說來，在 SCK-CEN 期間之工作模式大致可分成以下三個階段。

第一階段（1st 週）

首先閱讀與搜尋 BR3 計畫與有關除役工作之相關資料，閱讀內容包括分期計畫書、各階段執行期間之結案報告（~1994）、上網搜尋 1995 以後之執行資料與相關論文。

第二階段（2nd 週）

了解 BR3 之工作現況與主要工作項目後，即分別與各主要工作負責人相約進行細部討論。討論內容除了 BR3 除役工作之工法設計邏輯、機具等議題外；並對本所目前面臨之問題亦有著墨。在乏燃料運貯、水刀（HPWJ）、EDM Cutting、Plasma Cutting 與機械臂操作方面，亦跟隨負責人員進入實驗工場了解環境以了解工法設計之緣由。

相約討論之人員有：

Jérôme Dadoumont：Project Leader（underwater Plasma cutting & EDM cutting）

Yves Demeeulemeester：project engineer(品保作業、TRR 燃料池改善）

Pierre Valenduc：project engineer（自動化、TRR 燃料池改善）

Mathieu Ponnet：head of section decontamination（燃料池 2”管與鋁材去污問題）

Luc Denissen：project engineer HPWJC＋Robot（水刀切割原理）

Henry Davain：project engineer HPWJC＋Robot（水刀切割工法設計、機台結構與機械臂操作展示）

Luc Ooms： responsible fuel（BR3 乏燃料儲運、TRR 破損乏燃料討論）

Olivier Emond：responsible material flow（去污程序與物料流管理）

Vincent Massaut：Project Manager（計畫管理、計畫財務與水泥塊去污、TRR 燃料池問題改善與建議）

第三階段（3rd 週）

參加 BR3 計畫主辦為期一週（註）的訓練課程（圖二），訓練課程之主題包括：

1. Preparing a decommissioning project
2. Safety aspects in decommissioning
3. Regulatory aspects of decommissioning
4. Project management
5. Financial aspects of a decommissioning programme
6. Material management
7. Characterisation and measuring methods
8. Dismantling techniques
9. Decontamination techniques
10. Visit of BR3
11. Social Event

註：自費參加理論研習部份，€400

三、實習心得

在 SCK-CEN 實際參與 BR3 除役計畫，歷經了一個多月的實習過程所見所聞的心得如下：

1. BR3 電廠自 1962 年開始商業運轉成為比利時第一座 PWR 核電廠。在運轉期間 BR3 亦同時肩負著訓練鄰近國家 PWR 運轉人員之重責大任。1987 年 BR3 因為比利

時政府在安全之考量下停止運轉。總計 BR3 電廠共服役 25 年。1989 年政策決定開始執行 BR3 之除役計畫，於是 BR3 成為歐洲第一座進行除役之 PWR 核電廠。

2. BR3 核電廠自 1989 年開始執行除役計畫，迄今已完成大部份電廠組件之拆除。電廠內的乏燃料亦已於 2002 年 10 月完成裝罐（共 8 罐）並送至鄰近的 Belgoprocess 貯存廠存放。

3. BR3 的除役計畫主要分成三個主要階段：

①第一期計畫（1989~1991）

此一階段為 BR3 除役計畫技術之開發期。主要任務為除役工作之規劃與拆除技術之前導性研究。此一時期之工作大致可分為：

■系統除役工法之規劃

擬定出全廠各系統除役之先後次序、時程、人力需求、經費需求等

■切割工法設計與工法比較

整合歐洲學術界與工業界針對 BR3 電廠組件進行切割工法設計與工法比較。對 BR3 計畫而言，拆除所需之工法與機具之概念設計在此階段產生。

■切割機具設計與模擬測試

以 Plant-Specific 之觀念，針對 BR3 特有之環境與限制條件量身訂做專用拆除機械，並將欲拆除之組件以全尺寸模擬方式進行模擬切割試驗。開發之切割設備有：

Hydraulic Shears（圖三）

Reciprocating Saw（圖四）

Band Saw（圖五）

Circular Saw（圖六）

Internal Milling Machine（圖七）

EDM Milling Cutter（圖八）

High Pressure Water Jet Cutter（圖九）

Underwater Plasma Cutter（圖十）

Underwater 8-jaw fixture（圖十一）

與相關之夾治具與切割參數開發

■除污技術開發

主要著眼在軟式除污製程之開發（亦即 C.O.R.D. Process），其優點有：

平均除污係數要高（需大於 10）

最少的計量暴露率

置換出的污染物需呈單一型式

無需大量的設備投資

化學藥劑用量較少且較價廉

製程較簡單

吾需額外設備，吾需額外增設樹酯交換槽

操作容易且除污係數高

產生之廢樹酯數量較少

②第二期計畫（1992~1997）

主要針對 Primary Loop 之 Piping 及其組件與 RPV 內部，先進行去污使區域之輻射背景與污染程度降低至人員可接近進行拆除之程度。（圖十二）

除污工作方面：

完成 primary Loop（包括 Pipes、Pumps、Steam Generator、Pressurizer, etc）之去污，去污後 dose rate 為原來的 1/10

拆除工作方面：

1. 5.4 ton thermal shield 在水下由 EDM Milling

Cutter & Plasma arc 與機械切割之方式將之切割成3塊。在經過多次試驗（Cold Test）與數次的實際拆除工作後，BR3 人員逐漸採用以機械切割為主的工作方法，此一觀念之轉變主要肇因於非傳統切割方式在此一除役拆除工作中並未發揮其特長，反而有其他之副作用導致時間與成本的增加（如二次廢料、機械臂操作困難度較高、水屏蔽易污染使監視困難，圖十三）

2. Segmenting of Reactor Internals

主要有四 internals：（圖十四）

■thermal shield（圓桶狀不鏽鋼結構 $76.2^{t}\times 2432^{h}\times 5.5^{ton}$）：

水平切成 5 個環型圈（ring）。第一個環型圈在爐內完成切除，其餘 4 個環型圈吊至 refueling pool 再行切割成 segments。

■Lower Core Support Assembly

■Upper Core Support Assembly

■Reactor Vessel Collar and its Associated Instrumentation Basket（$\phi 1715\times 194^{t}\times 310^{h}$）

切割工法大多採用改良式之切割機具，切割機具都先由 BR3 人員完成概念設計並完成採購規範書之編撰，再由核能顧問公司如西門子、法馬通等公司承製並配合進行試俥。所有拆除工法均需進行全尺寸之試俥，且試俥時之成功率需達 95%以上，並在工作程序書完成審核後才准執行現場切割作業。

③第三期計畫（1998~2005）：RPV 本體切割與圍阻體拆除

RPV 整個移出其 cavity，在 refueling pool 進行切割。為切割 RPV 本體，BR3 工程師特別設計了 8 爪夾盤配合其他夾具與圓盤鋸、往覆鋸等特製工具在水底進行切割。為顧及切割時之安全與避免夾刀現象，有 2 台吊車吊持著 RPV 本體，並有吊索拉力與 RPV 姿態監視設備隨時與切割設備保持聯動。目前 RPV 切割仍持續進行中。

資料蒐集方面：共有

計畫書：3 份

- The BR3 Pressurised Water Reactor Pilot Dismantling Project Report EUR 18229en
- The BR3 Pressurised Water Reactor Pilot Dismantling Project Progress Report No.8
- RPV and Internals Dismantling Project Final Report, Aug 2000

SCK 內部工作程序書：1 份

Mechanical Decontamination of a Concrete Block with Diamond Disc

SCK 內部報告：5 份

- The BR3 Dismantling Operations and Related Techniques－Dismantling of the Reactor Pressure Vessel
- The BR3 Dismantling Operations and Related Techniques－Dismantling of Highly Radioactive Reactor Internals
- Waste Management at BR3
- Waste Management－Study of Dismantling Strategy
- Evacuation of the BR3 Spent Fuel

BR3 計畫產出之投稿論文：4 份

- Decontamination Strategy for the Dismantling of Strongly Contaminated Loops: the Practical Case of the Dismantling of the BR3 PWR Auxiliary and Primary Loops
- Overview of Recycling Technologies for Decommissioned Materials: Lessons Learned during the Dismantling of a Reactor
- Nuclear and Non-Nuclear Safety Aspects in Nuclear Facilities Dismantling: The Example of a PWR Pilot Decommissioning Project
- Dismantling of the BR3 Reactor Pressure Vessel

比利時拆除混凝土 free release 標準：1 份

除役受訓課程講義：1 份

4. 除役工作之經驗交流方面：

拆除技術方面：

BR3 計畫自 1987 年開始執行拆除計畫以來，曾在 1987~1989 年間針對除役拆除工法進行廣泛的工法研究。在這兩年的前導性計畫中，曾針對鋸切、銑削、鑽削等傳統式的機械加工方式與非傳統式的切割工法（Plasma Cutting、EDM、High Pressure Water Jet Cutting 等）進行研究。

工法研究之方式主要是先確定切割標的物後，再針對標的物所處之環境、材質特性、尺寸與幾何外型、輻射與污染之考量等因素選擇或設計工法。

各組件之切割工法確定後，需再完成機具設計並進行漫長的切割試驗。在此一階段 BR3 計畫下的 14 位工程師每人都至少負責 2 項工作方法的研究，且每一工程師彼此間又相互為彼此任務編組之組員。因此彼此間之主從關係並不明顯，加上平均年齡約

35 歲，因此比較能嚐試新的工作方法。但彼等在 BR3 計畫之平均年資小於 10 年且皆為 BR3 停止運轉後才加入除役工作，因此計畫主持人 Vincent Massaut 先生在計畫管理與人員訓練方面不餘遺力。雖然如此，BR3 計畫藉由與各核能顧問公司（法馬通、西門子、GNB 等）之合作關係，亦都自行完成許多機具之概念設計並逐步開發與整合出現場實用性極高之設備，且都一一完成了除役計畫所規畫之預期目標。

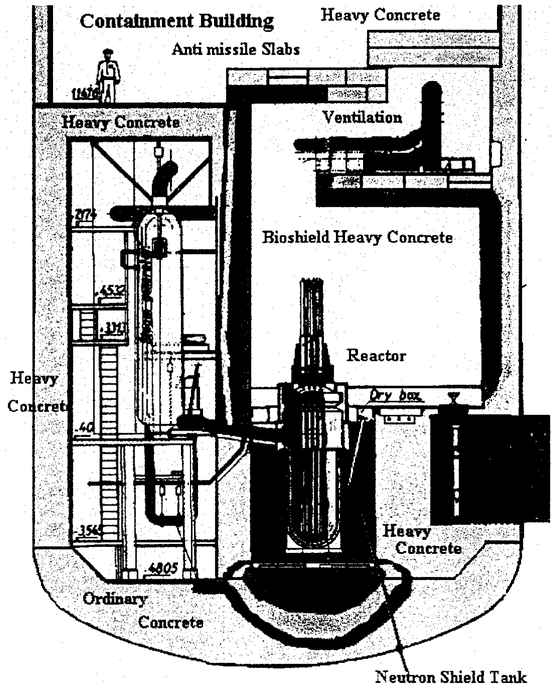
在除役拆除作業方面，依據十餘年之工作經驗，BR3 計畫有以下的心得；其認為若採用濕式切割工法，大可在反應器停止運轉後隨即展開拆除作業，因為對於 0 waiting time、10-year waiting time 或更久，所涉及之輻射強度衰減的考量，可輕易的由增加數公尺的水屏蔽厚度即可解決，且是否有足夠之等待期對機具或工法之選擇方面並無太大之差異。因此 Vincent Massaut 先生建議無需長時間等待反應器或其他組件自然衰減其活度。而只要除役規劃完成且人員與機具準備就緒即可進行拆除之工作。對於除役機具之選擇，則建議優先考慮以傳統式的切割機具，並依除役現場狀況進行改良與整合。採用何種工法或機具並無一定論，完全依除役現場環境、人員劑量曝露與二次廢料產生量與其他方面之特殊考慮來做最適當之組合。這部份工作實有賴對除役設施與拆除作業都有深刻了解之人士進行設計才能收事半功倍之效。

TRR 燃料池清理之困難：

對於 TRR 燃料池處理之困難，藉由身處歐洲核能工業重鎮的 SCK；同時亦藉著參加除役訓練課程之便；職曾與 BR3 同仁及一同參加受訓之核能同業詢問有關於本所用過燃料池處理方面的建議性作法。為增進彼等對 TRR 燃料池問題之了解，職曾私下以相片與影片詳細解說。雖然彼等都有相當豐富之工作經驗，但在觀看了影片後都一致認為問題實在棘手，因此在一時間都無法提出有效的解決方案。

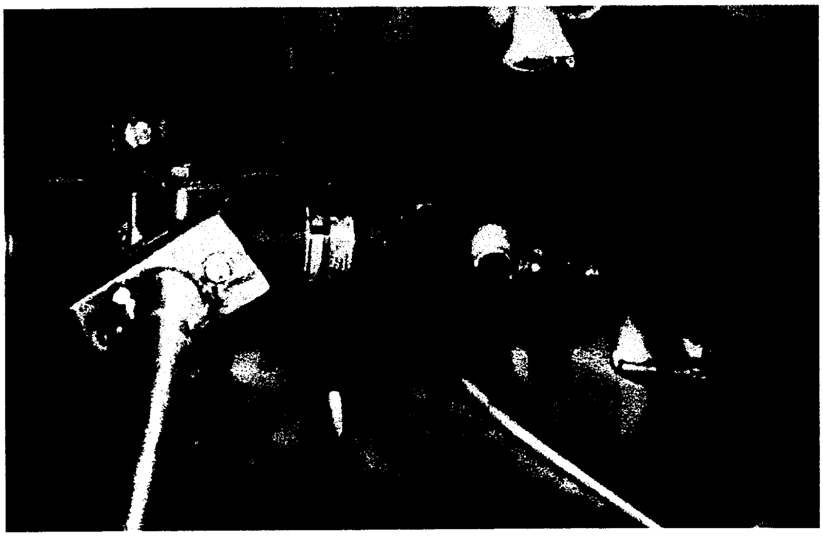
四、建議

1. 因應除役工作為一長期之計畫，應及早培訓除役工作人才。
2. 加強與歐洲研究機構之交流
3. TRR 拆除與遷移之經驗與工法值得投稿
4. 建議另派員參與 2003 年之除役訓練

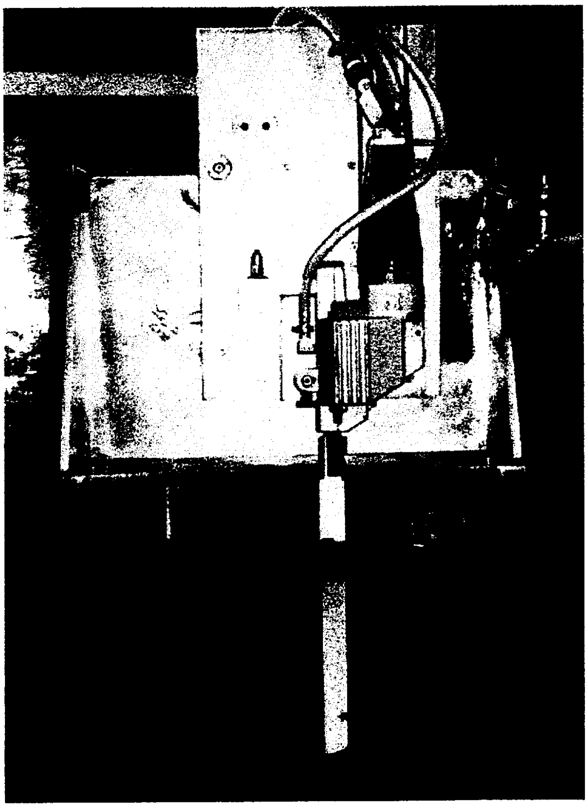


圖一：BR3 電場廠內系統示意圖

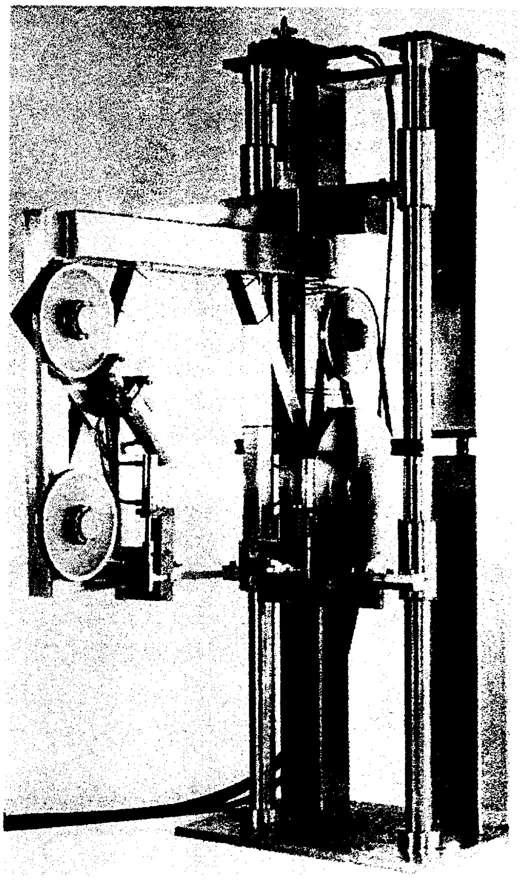
圖二：參與除役課程講習學員留影



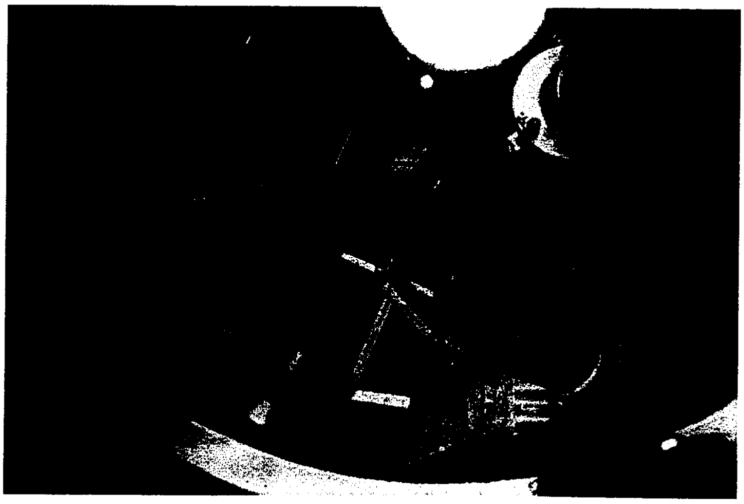
圖三：Hydraulic Shear



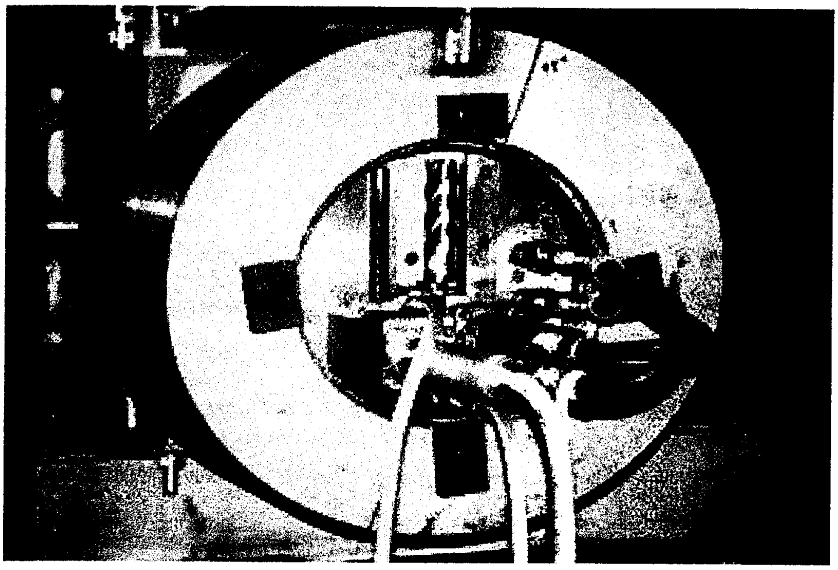
圖四：Reciprocating Saw



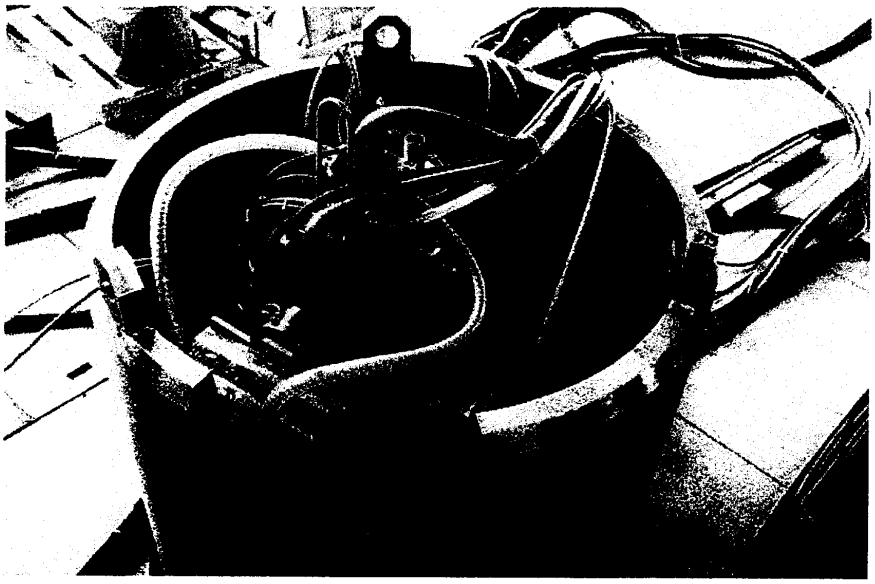
圖五：Band Saw



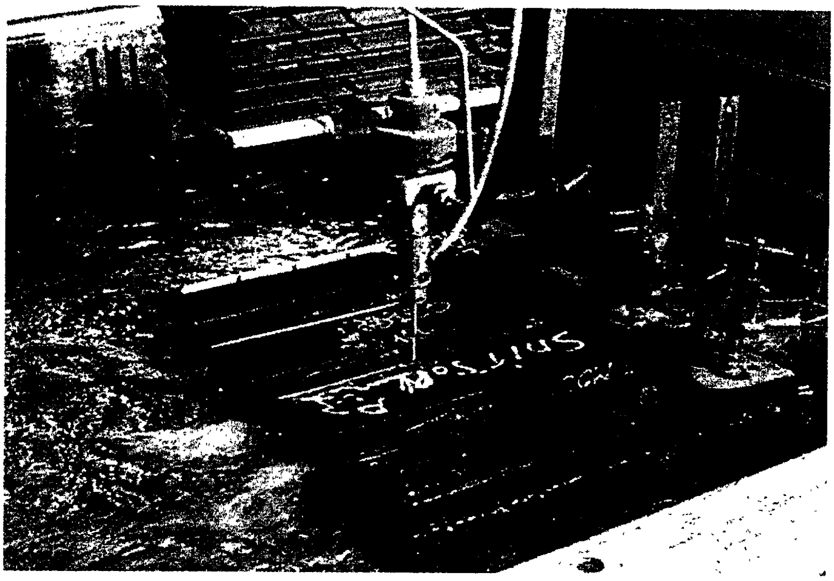
圖六：Circular Saw



圖七：Internal Milling Machine



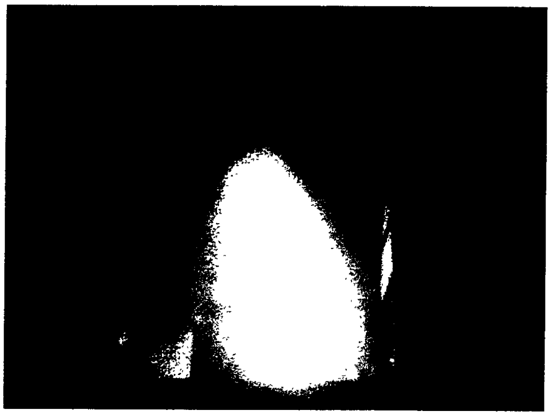
圖八：EDM Milling Cutter

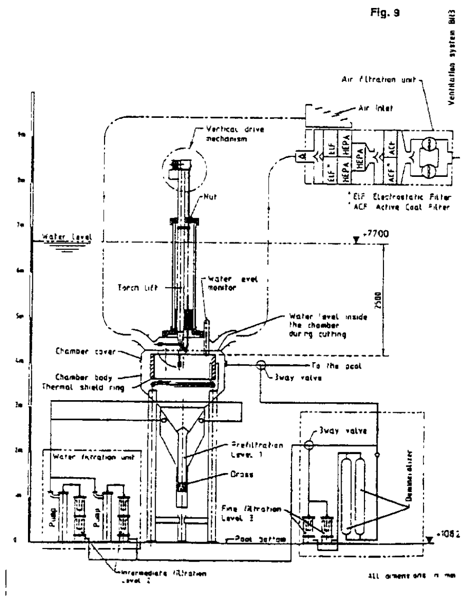


圖九：High Pressure Water Jet Cutter

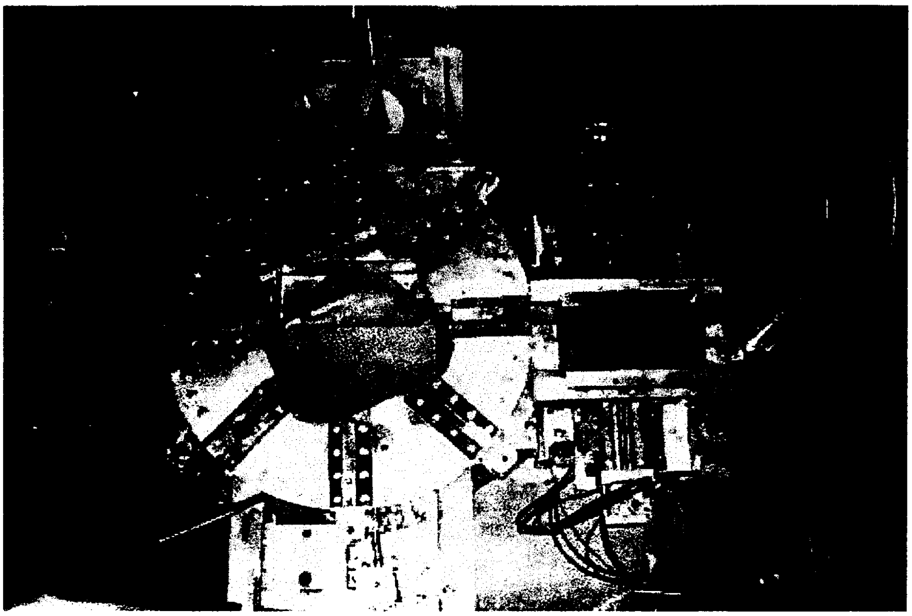
Flooded chamber for plasma arc torch cutting

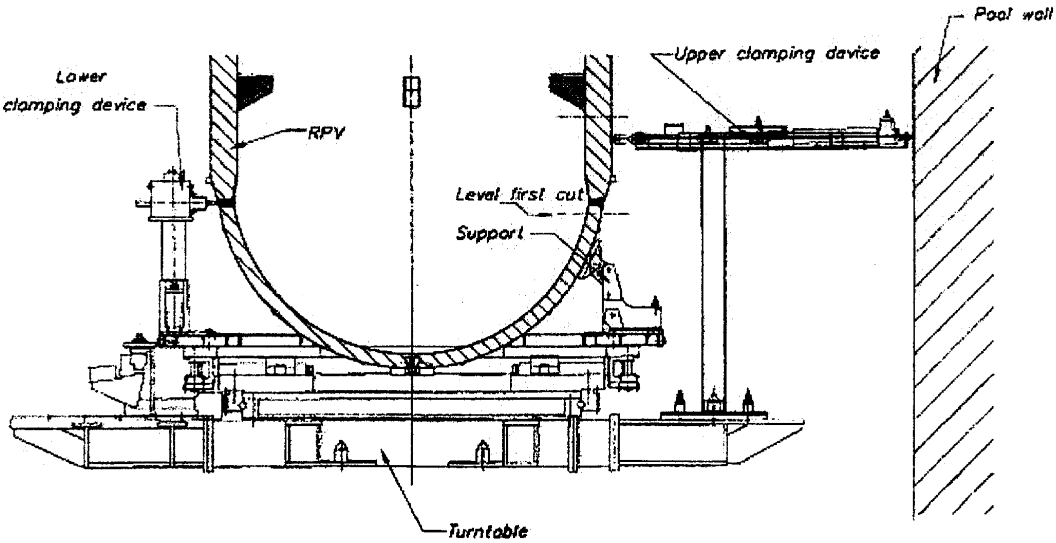
Fig. 9



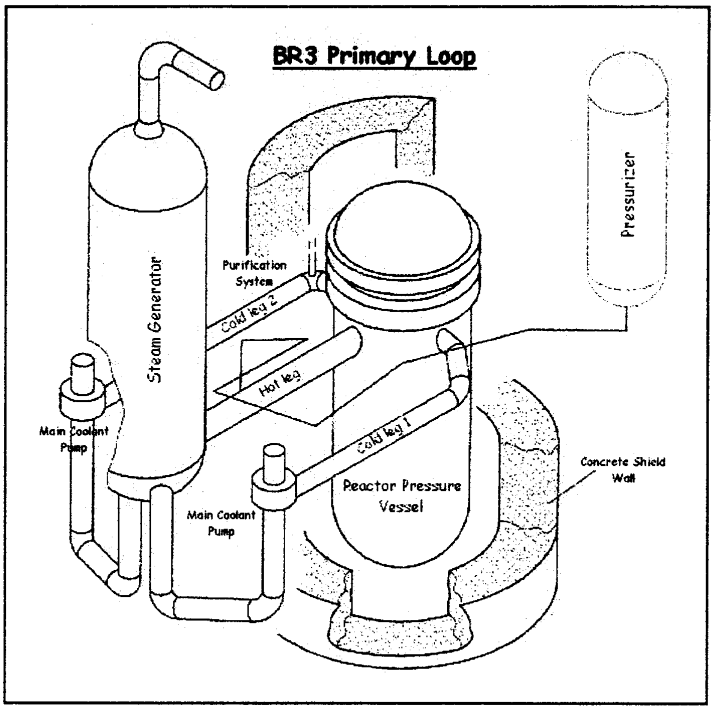


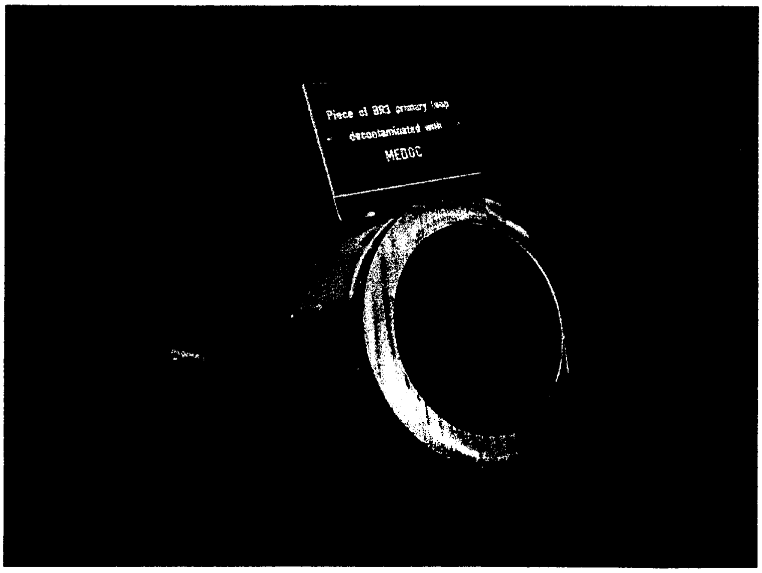
圖十：Underwater Plasma Cutter





圖十一：8-Jaw Underwater Fixture





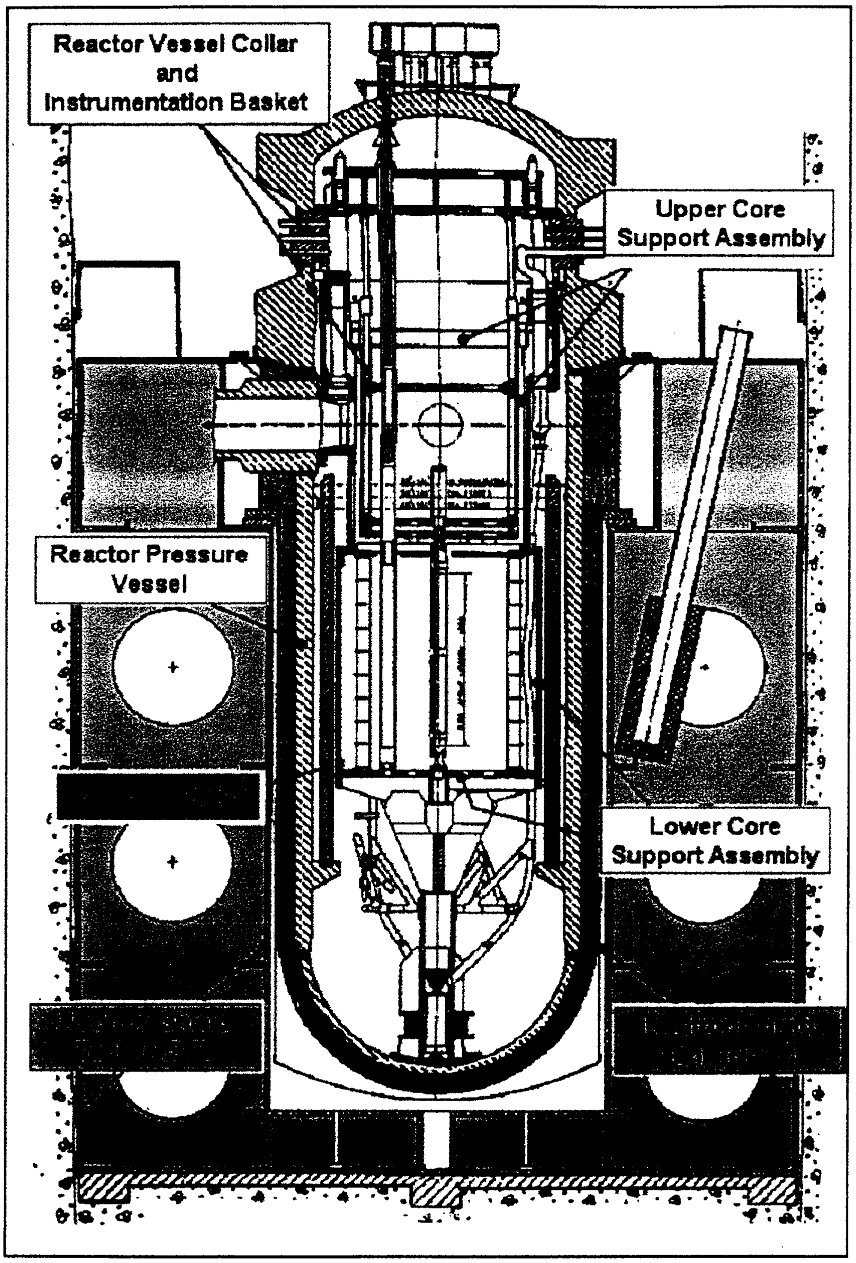
圖十二：Primary Loop Pipes Decon

Cutting method	Cut length	Kerf width / depth Nº x Length	Cutting speed (1)	Time for cutting + preparation	Effective cutting speed	Metal removed from thermal shield	2^{nd}ary waste volume	Ratio waste vol/cut length (1)	Dose uptake (6)	Dose uptake per cut length
	[m]	[mm]	[mm/m]	[hrs]	[mm/m]	[dm^3]	[dm^3]	[dm^3/m]	[mSv]	[mSv/m]
Vertical EDM	2.92	7/76.2 8 x 365	0.6	184	0.27	1.622	96 (2)	33	7.5	2.56
Horizontal Mechanical	4.15	4/18-20 4 x 4150	6	64	1.08	1.262	26.7 (3)	6.4		
Horizontal Mechanical	4.15	4/18-20 4 x 4150	6	80	0.88	1.262	26.7 (3)	6.4	8.3	0.66
Horizontal Mechanical	4.15	4/18-20 4 x 4150	6	48	1.44	1.262	26.7 (3)	6.4		
Horizontal EDM	4.15	7/76.2 12 x 346	0.6	240	0.29	2.208	128 (2)	31	7.1	1.71
Vertical Plasma	14.98	11/76.2 24 x 482 8 x 426	300	136	1.83	12.55	459 (4) (5)	31	9.2	0.61

- For 76.2 mm wall thickness. For EDM, this corresponds to an equivalent speed, the way of cutting being different from the two other methods
- Fine filters containing activated EDM particles
- Coarse filters containing activated chips
- Coarse, intermediate & fine filters; demineralizer columns not completely saturated
- Not taking into account the waste produced by the air filtration system
- The preparatory phase, including the unloading of the reactor internals gave rise to a total dose uptake of 22.6 man-mSv

Table III : Thermal Shield Segmentation

圖十三：各式工法切割 Thermal Shield 之比較(Cold Test)



圖十四：Segmenting of Reactor Internals

附件

一、BR3 拆除計畫計畫書：3 份

- The BR3 Pressurised Water Reactor Pilot Dismantling Project Report EUR 18229en
- The BR3 Pressurised Water Reactor Pilot Dismantling Project Progress Report No.8
- RPV and Internals Dismantling Project Final Report, Aug 2000

二、SCK 內部工作程序書：1 份

Mechanical Decontamination of a Concrete Block with Diamond Disc

三、SCK 內部報告：5 份

- The BR3 Dismantling Operations and Related Techniques －Dismantling of the Reactor Pressure Vessel
- The BR3 Dismantling Operations and Related Techniques －Dismantling of Highly Radioactive Reactor Internals
- Waste Management at BR3
- Waste Management－Study of Dismantling Strategy
- Evacuation of the BR3 Spent Fuel

四、BR3 計畫產出之投稿論文：4 份

- Decontamination Strategy for the Dismantling of Strongly Contaminated Loops: the Practical Case of the Dismantling of the BR3 PWR Auxiliary and Primary Loops
- Overview of Recycling Technologies for Decommissioned Materials: Lessons Learned during the Dismantling of a Reactor
- Nuclear and Non-Nuclear Safety Aspects in Nuclear Facilities Dismantling: The Example of a PWR Pilot Decommissioning Project
- Dismantling of the BR3 Reactor Pressure Vessel

五、比利時拆除混凝土 free release 標準：1 份

六、除役受訓課程講義：1 份

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ISSN 1018-5593

European Commission

nuclear science and technology

The BR3 pressurised water reactor pilot dismantling project

Report

EUR 18229 EN

EUROPEAN COMMISSION

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responsible for research, innovation, education, training and youth

DG XII/F.5 — R & D programme: 'Decommissioning of nuclear installations 1989-93'

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European Commission

nuclear science and technology

The BR3 pressurised water reactor pilot dismantling project

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Contract No FI2D-CT89-0003

Final report

This work was performed in the framework of the European Atomic Energy Community's research programme on the decommissioning of nuclear installations (1989-93)
Area C: 'Testing of new techniques in practice'

Directorate-General
Science, Research and Development

1998 EUR 18229 EN

A great deal of additional information on the European Union is available on the Internet. It can be accessed through the Europa server (http://europa.eu.int).

Cataloguing data can be found at the end of this publication.

Luxembourg: Office for Official Publications of the European Communities, 1998

ISBN 92-828-3615-0

Printed in Luxembourg

PRINTED ON WHITE CHLORINE-FREE PAPER

Table of contents

Abstract

The BR3 Pressurized Water Reactor was started in 1962 and definitely shut down in 1987. Selected in 1989 by the European Commission as one of the four pilot decommissioning projects in the framework of the EC five-year RTD programme on the decommissioning of nuclear installations, the project started on site operation in 1991 with the full system decontamination of the primary loop.

Afterwards, the reactor thermal shield was dismantled, allowing to compare three main underwater cutting techniques : the plasma arc torch, the electric discharge machining and the mechanical cutting.

The next step was the remote dismantling of the remaining reactor internals, using the lessons drawn during the first phase. Most of the dismantling was carried out with mechanical cutting methods.

Finally, a second set of internals, unloaded from the reactor in 1964 and stored since then in the reactor refuelling pool was also dismantled. This allowed to compare the operation on pieces with a cooling down period of 30 years with the dismantling carried out shortly after shutdown. The waste management and evacution is also described in this report.

Acknowledgements

This pilot project has been performed with financial support of the European Commission Directorate-General XII, in the framework of its five-year RTD programme on the Decommissioning of Nuclear Installations.
Throughout the whole project duration, the Commission Delegates have helped the Dismantling Team by fruitful discussions in small committees and in large international meetings.

The project could only be successfully realized thanks to the enthusiastic involvement of personnel of SCK•CEN from the board of administrators to the executing technicians. In particular, it is worthwile to mention the active participation of Guy Collard, Jérôme Dadoumont, Luc Noynaert, Marc Vankeerberghen, Marc Van Broekhoven, Pierre Verjans, Robert Mandoki, Yves Demeulemeester, Charles Plateau, Roger Henderix, Jan Swinnen, the whole personnel of the BR3 project and the personnel of various services of SCK•CEN also involved in the project.

The team is especially thankful to Mr. François Motte. He did switch off the BR3 reactor in June 1987 after 25 years of operation, initiated in 1989 the Dismantling Project and headed the Dismantling Team till its retirement in 1992.

We may not forget the strong support and the enlightened advises of Prof. Luc Gillon who, after a long career in the Nuclear World, has still the enthusiasm of a young engineer and helped the team to surmount some of the difficulties encountered during the project.

The demonstration effect of the pilot project and its international recognition could only be achieved by the active participation of the associated contractors, the international and national partners and the industries involved in the realization of the Dismantling Programme. We mention in particular Mr. Alain Lefèbvre, delegate of Belgatom, who actively participated to the project since its beginning.

In conclusion, we want to thank all the persons who participated to the successful realization of this pilot project, regardless of their position. We are sure that they will continue the further dismantling of the BR3 reactor with the same enthusiasm and technical skills.

Executive Summary

The BR3 reactor was the first PWR installed in Europe. It is a quite small reactor, with an electrical net power output of 10.5 MWe, but presenting all the main features of commercial PWR plants.
Started in 1962, the reactor was definitely shut down in June 1987, after 25 years of operation. In 1989, the BR3 was selected by the European Commission as one of the four pilot decommissioning projects in the framework of its five-year RTD programme on the decommissioning nuclear installations.
The contract signed with the Commission was first divided into two phases, covering the years 1989-1991 and 1992-1994, and then an extension of the contract was agreed covering the activities performed mostly in 1995. This report presents the results obtained during this 6 year-contract.

The different phases of the contract covered various activities which will be highlighted in the report :

- *Phase 1 (1989-1991) :*
  - *pre-dismantling full system chemical decontamination of the primary loop*
  - *comparison of cutting techniques for the dismantling of highly radioactive internals, and*
  - *testing of these techniques on a first reactor internal, the thermal shield*
- *Phase 2 (1992-1994) :*
  - *dismantling of the remaining reactor internals, using the lessons learned during the preceding phase*
- *Phase 2bis (1994-1995):*
  - *dismantling of the first set of reactor internals having undergone 30 years cooling down period and comparison with the direct dismantling of the other internals.*

These different operations have brought a lot of information which will be summarized in the present report. It concerns mostly the summary of the selection process, the results and the usefulness of the primary loop decontamination, as well as the results and lessons drawn from the dismantling of the highly radioactive reactor internals.
Comparison of remote underwater cutting and dismantling techniques for highly radioactive internals is reported; regarding the shape and materials of some specific internal pieces some results can also be applied to the cutting of reactor pressure vessels. This report gives also data on operation duration, manpower, distributed doses and generated waste.

This report is a common product of SCK•CEN (Belgian Nuclear Research Centre) which owns the reactor and acted as principal contractor, Siemens/KWU (Germany), Framatome (France), Rolls-Royce and Associates Ltd (U.K.) which were associated contractors to the contract and finally Belgatom (Belgium), partner in the project.

1. INTRODUCTION : Short history of the BR3 reactor

The BR3 reactor (**B**elgian **R**eactor No. **3**) was the first PWR ordered and connected to the grid in Western Europe. It was ordered to the Westinghouse company (US) in 1956 and started operation in October 1962. The reactor had a thermal power generation of 40.9 MWth, a gross electricity output of 11.5 MWe for a net power output of 10.5 MWe.
The BR3 was mainly used for training commercial reactor operators and for testing advanced fuels (high burnup, burnable poison, MOX fuels) in full PWR conditions.
During its lifetime, starting in 1962 and ending on 30th June 1987, the BR3 produced 964.3 GWh of electricity in 11 operating campaigns (see table I) and underwent three main operations :

- In 1964 the reactor internals were removed and exchanged by new ones for carrying out an experiment called "Vulcain". This experiment involving a mixture of heavy and light water required also some changes to the auxiliary loops in order to control the mixture composition and to recover the heavy water.
- In 1975 the primary loop was decontaminated by a chemical process called Turco.
- Finally, in 1984, one carried out a wet annealing of the pressure vessel to decrease the neutron induced embrittlement of the RPV material and thus to allow further operation of the plant.

On June 30th, 1987 at 24.00 hrs, the reactor was definitely shut down.

Table I : THE ELEVEN BR3 OPERATING CAMPAIGNS (1962-1987)

Operation period	Electrical Energy in millions of kWh Gross (produced)	Electrical Energy in millions of kWh Net (provided to the grid)	EFPH	Availability factor of the plant, %	Denomination of the successive cores
10.10.62 up to 21.08.63	45.8	40.8	4008	62	Core BR3/1A
02.12.63 up to 31.07.64	55.5	51.1	4848	90	Core BR3/1B
29.11.66 up to 18.11.68	159.9	142.9	13944	90	Core BR3/2
31.07.69 up to 20.12.70	79.2	67.1	7339	91	Core BR3/2B
25.09.72 up to 11.01.74	89.4	78.7	7944	80	Core BR3/3A
02.07.74 up to 27.06.75	47.9	40.2	4416	76	Core BR3/3B
15.07.76 up to 15.04.78	132	117.1	11916	96	Core BR3/4A
22.06.79 up to 26.09.80	97.5	87.2	8663	86	Core BR3/4B
21.09.81 up to 01.04.83	98	86.3	8641	90	Core BR3/4C
13.07.84 up to 11.11.85	89.5	79.2	8008	85	Core BR3/$4D_1$
03.07.86 up to 30.06.87	69.6	61.2	6232	92	Core BR3/$4D_2$
TOTAL	964.3	851.8	85959		

2. TIME SCHEDULE AND PHASES OF THE PILOT PROJECT

The BR3 decommissioning was selected in 1989 by the European Commission as one of the four pilot projects in the framework of its five-year RTD programme on the decommissioning of nuclear installations. The first contract was signed in October 1989 with an effective commencement date on 1st October 1989.

The operations were divided into two main phases, as requested by the Commission.
The **first phase** comprised mainly the carrying out of a pre-dismantling full system decontamination of the primary loop, the selection and testing of cutting techniques for the dismantlement of highly radioactive internals and then the actual dismantling of a first reactor internal, the thermal shield.
The **second phase** involved the dismantling of all the remaining reactor internals.
At the end of the contract, an extension was signed which involved the dismantling of a first set of internal pieces, unloaded in 1964, and the comparison of this operation (concerning pieces with a cooling down period of 30 years) with the preceding one, done almost immediately after shutdown. This phase was called the **phase 2bis** of the pilot project.

These phases of the project were broken down into working packages as follows :
Phase 1 of the contract :

- B.1. Decontamination of the reactor primary loop
  - B.1.1. Cost benefit analysis and selection of a procedure
  - B.1.2. Decontamination operation
  - B.1.3. Treatment and removal of decontamination waste
- B.2. Dismantling of internals
  - B.2.1. Concept and design of the segmenting and remote-operating equipment
  - B.2.2. Manufacturing and procurement of the segmenting and remote-operating equipment
  - B.2.3. Inactive testing and commissioning of the segmenting and remote operating equipment
  - B.2.4. Segmenting of activated components
  - B.2.5. Waste treatment and packaging
- B.3. Generation of specific data

Phase 2 of the contract :

- B.3.ext. Collection of specific data (extension)
- B.4. Disassembling of the main reactor internals
  - B.4.1. Lower Core Support Assembly
  - B.4.2. Upper Core Support Assembly
  - B.4.3. Reactor Vessel Collar and Instrumentation Basket
- B.5. Segmenting of stainless steel reactor internals subassemblies
  - B.5.1. Annular cylindrical geometries (3D geometries)
  - B.5.2. Plates and grids (2D geometries)
- B.6. Segmenting of a thick carbon steel ring (Reactor Vessel Collar)

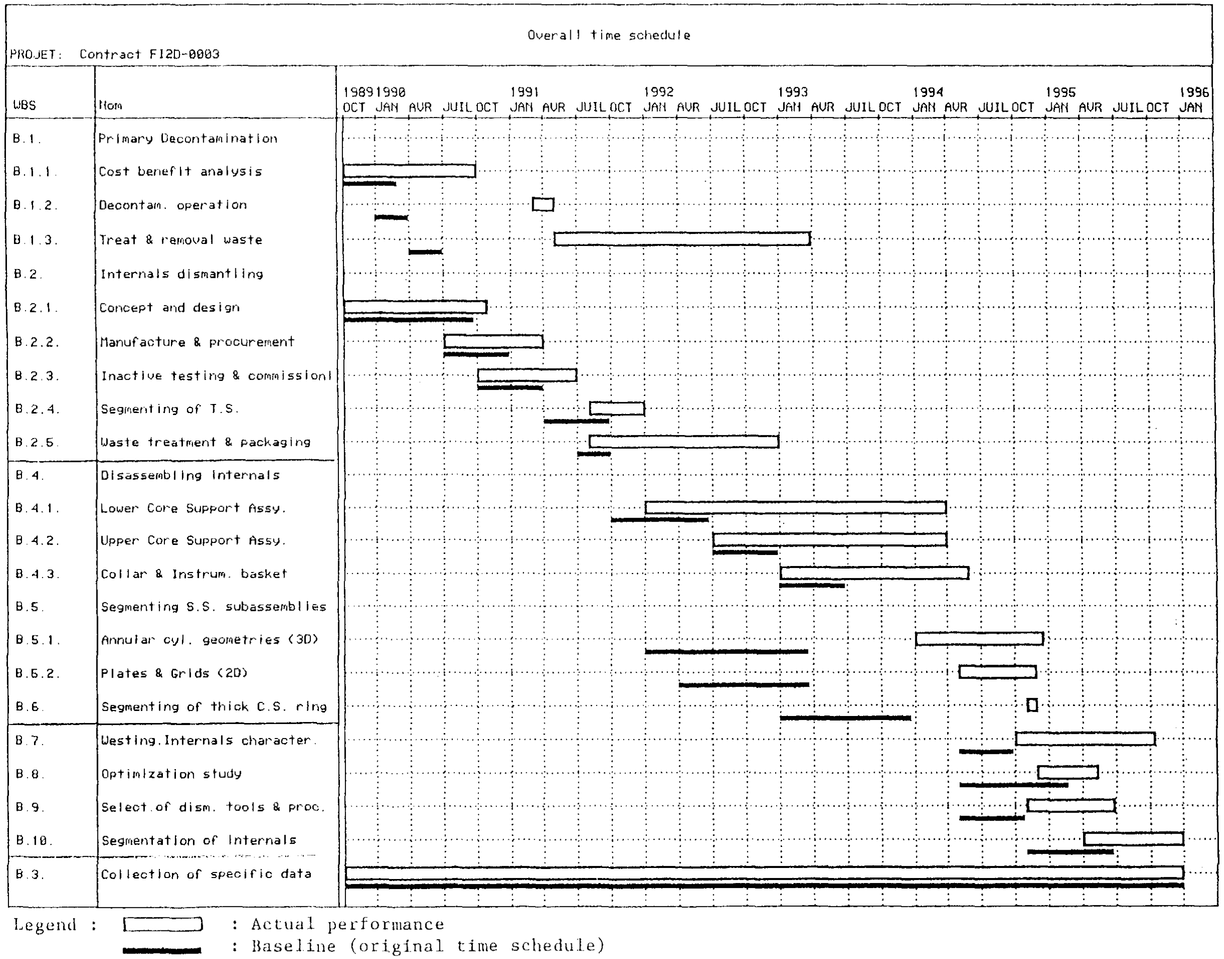
Phase 2bis of the contract :

B.3.ext. Collection of specific data (extension)
B.7. Internals
B.8. Optimization study - ALARA and techniques
B.9. Selection of dismantling tools and procedures
B.10. Segmentation of Westinghouse internals

The original and present time schedules are given in the bar chart hereafter. The operation titles refer to the working package definition.
One can note that the waste management and evacuation was not part of the contract but has implication on the actual realized time schedule.

As far as the associated contractors are concerned, Siemens KWU and Framatome were involved in phase 1 of the project, while Rolls-Royce and Associates Ltd was involved in phase 2.

Further on in this report, a different presentation of the activities carried out and of the gained results is used, for better clarity and systematic presentation. Nevertheless reference to the working package will be made.


Overall time schedule
PROJET: Contract FI2D-0003
WBS
Nom
1989 1990 1991 1992 1993 1994 1995 1996
OCT JAN AVR JUIL OCT JAN AVR JUIL OCT JAN AVR JUIL OCT JAN AVR JUIL OCT JAN AVR JUIL OCT JAN AVR JUIL OCT JAN
B.1. Primary Decontamination
B.1.1. Cost benefit analysis
B.1.2. Decontam. operation
B.1.3. Treat & removal waste
B.2. Internals dismantling
B.2.1. Concept and design
B.2.2. Manufacture & procurement
B.2.3. Inactive testing & commissionl
B.2.4. Segmenting of T.S.
B.2.5. Waste treatment & packaging
B.4. Disassembling internals
B.4.1. Lower Core Support Assy.
B.4.2. Upper Core Support Assy.
B.4.3. Collar & Instrum. basket
B.5. Segmenting S.S. subassemblies
B.5.1. Annular cyl. geometries (3D)
B.5.2. Plates & Grids (2D)
B.6. Segmenting of thick C.S. ring
B.7. Westing.Internals character.
B.8. Optimization study
B.9. Select.of dism. tools & proc.
B.10. Segmentation of internals
B.3. Collection of specific data
Legend :
: Actual performance
: Baseline (original time schedule)

3. PRE-DISMANTLING FULL SYSTEM DECONTAMINATION OF THE BR3 PRIMARY LOOP

3.1. Choice of the decontamination process

The Full System Decontamination of the primary loop and of the associated circuits can only be performed by a chemical process. Moreover, it is not recommended to postpone for too long this operation: this implies indeed that the loop in its entirety as well as its active components like the primary pumps, are in a full satisfactory operational condition. At longer term, it is difficult and costly to guarantee the maintaining of this condition. Moreover, the presence of qualified operators for running the loop can only be guaranteed during a limited time after the reactor stop.

The decontamination operation will allow the further dismantling in safer conditions thanks to the **dose reduction** and thanks to the **removal of the contaminated crud** so that the spreading of the surface contaminants will be greatly reduced during the cutting operations.

The most important **specifications** for the full system chemical decontamination were defined as follows:

1. to reach a high mean Decontamination Factor, at least a DF of the order of 10 for the primary loop;
2. to minimize the radiation dose associated with the operation;
3. to minimize the secondary waste production by concentration of the removed corrosion products and the associated radioactive contamination in a single form, active ion exchange resins, with a volume as low as possible;
4. to use the plant as it is, without expensive additional equipments or modifications.

Criteria 1, 3 and 4 are directly related to the choice of the decon process; criterion 2 is very important and depends not only on the choice of the process but also on a good preparation of the operation itself and the treatment of the waste generated by the decon operation.

An important fraction of the radiation dose distributed among the reactor operators is related to unavoidable operations, upstream and downstream from the decon operation itself : more particularly, for the typical case of the BR3 reactor, the closing and opening of the reactor cover which is still performed manually.

A good preparation and a careful maintenance of the equipments before doing the decon operation are essential to reduce the risks of incidents (leaks of decon solutions) during the operation and the related risks of radiation dose associated to such eventual incidents.

3.2. Description of the process

3.2.1. Choice of the process

The decontamination processes used nowadays in operating plants are mainly "soft processes" i.e. processes working at low concentration of chemicals; the so-called "hard processes" working at high concentration of chemicals are less used because they generally lead to the formation of high amount of solid waste. It must be noted, however, that the BR3 plant was decontaminated in 1975 with a "hard process", the so-called "Turco process".

A comparison was made between 3 groups of soft processes (the Candecon, Canderem, dilute Citrox processes; the LOMI process; the MOPAC 88 and CORD processes).
Thanks to the very low amount of salts introduced in the system, the CORD process leads to the lowest waste production. Moreover, it requires only a few modifications to the plant and is easier to operate (e.g. the LOMI process requires the use of a nitrogen atmosphere). For these reasons, the CORD process developed by Siemens KWU was finally selected for the full system decontamination operation.

3.2.2. Description of the CORD process of Siemens KWU

CORD is an acronym for Chemical Oxidizing Reducing Decontamination. It is a proprietary process of SIEMENS AG KWU.
For PWR oxide layers of the Cr and Ni bearing Fe_3O_4 type, an oxidizing pretreatment is always necessary. This pretreatment (**OXIDATION STEP**) allows the oxidation of Cr^{3+} into soluble Cr^{6+}. Afterwards, the iron oxides with the contaminating radionuclides are dissolved (**DECONTAMINATION STEP**) by organic acids/complexing agents such as oxalic acid. To prevent redeposition of the metal ions, a chelating agent is used to increase the solubility of the metal ions in solution by forming a chelating complex. Finally, the excess organic acid is decomposed by oxidation (**CLEANING STEP**).
One ***DECON CYCLE*** comprises the three steps ***OXIDATION, DECONTAMINATION and CLEANING.*** To remove the crud layer as far as possible and so to reach a high decontamination factor, it is necessary to repeat these decontamination cycles several times. For the BR3 reactor, it was decided after laboratory tests on BR3 representative test pieces to perform three cycles.

The oxidation step is performed at 100°C with $HMnO_4$ at a concentration of 0.3 g/l. The decontamination step is performed between 85°C and 100°C with oxalic acid at a concentration of 3 g/l. The cleaning step is performed using permanganic acid for the first two cycles and by oxidation with H_2O_2 on a catalytic bed for the third cycle.
The cationic species (Mn^{2+}, Ni^{2+}, and Co^{2+}) are removed by cation-exchange columns; the anionic species (iron and chromium oxalates) are removed on anion-exchange columns. The catalytic bed for the oxalic acid destruction is an anion-exchange resin with Pd as catalyst; the catalytic bed is operated at a temperature of 85 °C which requires the use of a heater upstream of the column and a cooler downstream.

3.3. Description of the BR3 system and of the modifications for the decontamination

The main system to be decontaminated was the reactor primary loop; figure 1 gives a schematic view of this loop; it comprises only one steam generator but, for obvious safety reasons, two primary pumps giving this so-called "1.5 loop system" configuration with two cold legs and one hot leg. The decontamination was performed with the fuel being unloaded. The total water inventory of the loop is of the order of 15 m^3; it forms the high pressure part of the plant.

The Purification System (PU loop) is associated to the decon process. Two important components of the Residual Heat Removal system (RHR loop) were also treated by the decon solutions; these were the Shutdown Cooling Heat Exchanger and the Emergency Shutdown Condensor together with their associated piping connecting them to the PU loop.

The total **surface to decontaminate was about 1200 m^2.** Most of the material in contact with the chemicals is SS 304, smaller parts are in SS 316 (9 m^2), in carbon steel (15 m^2), in chromium plated SS (10m^2) and in zircaloy-4 (5.5 m^2).

3.3.1. Operating working parameters

The required temperature is achieved by the heat losses of the 2 primary pumps. When the operating temperature of 80°C to 100°C is reached, the heat input by the pumps becomes too high; heat is withdrawn from the system by the purification system heat exchanger and mainly by using the Steam Generator as a liquid-liquid heat exchanger. Therefore, some modifications had to be done to the secondary side of the SG to establish a closed water cooling secondary loop itself cooled by tertiary water from the lagoon.

To avoid cavitation problems at the suction head of the primary pumps, it was necessary to operate the primary circuit at a pressure higher than 20 bars. Therefore the Pressure Relief Safety Valve of the Pressurizer has been replaced to reduce the opening pressure. The water level in the pressurizer is controlled by water feed and bleed from the purification circuit and by a nitrogen feed in the pressurizer.

3.3.2. Ion exchange columns

Based on the estimation of the corrosion product inventory, the total resin volume needed was estimated. It appeared that the installed capacity of the BR3 plant (3 columns of 212 l capacity each) was too small so that 3 additional columns (100 l capacity each) of the mobile Siemens equipment AMDA were integrated in the purification system. The columns were installed and interconnected so that it was always possible to by-pass the columns or to combine cationic and anionic columns to meet the requirements of the process and to exhaust the resins completely.

3.3.3. Injection and dosing of the chemicals

A chemical injection and preparation skid of the mobile AMDA equipment was installed in the shipping area of the plant. It comprised a heated mixing tank for the preparation of the solutions and injection pumps for feeding the solutions in the Pu surge tank of the purification system.

3.4. Decontamination operation

The decontamination process was carried out from April 9 at 07.00 a.m. to April 18 at 07.00 a.m., extending over a period of exactly 9 days of continuous operation. No incident occurred and the primary leak rate remained negligible as it was at the start of the operation. Maintenance was required on one of the two so-called Charging and Circulating Pumps of the purification system: they are piston pumps and the piston surface is chromated. Attack of the chromated layer by the decon chemicals led to a loss of efficiency of the pump and to an excessive leak rate so that the pistons had to be exchanged for spare pistons.

3.4.1. Release of activity and corrosion products

During the 3 CORD cycles, a total of 1981 GBq (53.5 Ci) was released. Figure 2 illustrates the released activity per decontamination cycle.
The gamma spectrometry analysis of liquid samples showed that the dominating nuclide is **^{60}Co**.
The mean value for the three cycles is as follows :

Co^{60} : 99 % ; **Mn^{54}** : 0.45 % ; **Cs^{137}** : 0.2 % ; **Sb^{125}** : 0.35%

The released alpha activity amounted to 2331 MBq among which 184.5 mg Pu.
The release of corrosion products present in the crud layer is illustrated in figure 3.
A total of 23.72 kg of Fe, Cr and Ni was dissolved corresponding to about 33 kg of oxides.
Some observations can be made:

- During the oxidation phase, only Cr is dissolved and there is nearly no activity release.
- During the decontamination phase, Fe and Ni are rapidly dissolved and the activity release is also quite rapid.
- Cycle 2 showed a particular behaviour: during the decontamination step, two peaks of corrosion products release and activity were observed. The first one at the beginning of the decontamination step and the second one about 20 hours later. Moreover, the examination of test samples placed in the primary fluid revealed the appearance of a yellow deposit on the pieces which could correspond to a deposit of ferrous oxalate.
- The total amount of material dissolved corresponds to a removal of about 2 mg/cm^2 (total surface 1200 m^2) assuming a uniform crud deposit which is probably not valid.
- The mean activity removed amounts to 170 kBq/cm^2 which is higher than the value of 40 to 50 kBq/cm^2 measured on test pieces from the reactor. This confirms the presence of deposits in some regions and the non-uniformity of the crud thickness.

3.4.2. Release and fixation of activity and corrosion products on ion exchange resin columns

During the decontamination step in each cycle, first the excess permanganic acid is reduced liberating free cationic Mn^{2+}. Then during the first hours of the decontamination step, most of the activity and of the corrosion products are dissolved liberating cationic species (Ni^{2+}, Mn^{2+}, Co^{2+}) and anionic species (iron and chromium oxalates and free oxalate). When the reduction reaction of permanganic acid with oxalic acid is finished, then the primary solution is sent on the ion exchange columns. Generally, one cationic and one anionic column are used in series.
The purification flow rate is quite low (maximum 4 m^3/h for a total primary solution volume of 15 m^3) so that about 24 hours are needed to purify the solution.

3.4.3. Production of wastes

The main solid waste generated were exhausted anionic and cationic resins. About 33 kg of corrosion products and 1981 GBq (53.54 Ci) were removed so that 1.37 m^3 of resin was finally used (524 l high active cationic resins, 632 l anionic and 214 l mixed bed both medium active wastes).

The ion exchange resins have been conditioned by embedding into cement in a 400 l drum. The high active resins were conditioned at the WAB installation of the Doel reactor plant and the medium active were conditioned at the BELGOPROCESS plant in Dessel. At Belgoprocess, the resins were mixed with cement and contaminated concrete whereas at Doel only cement was used. The total amount of conditioned waste was 19 drums of 400 l corresponding to a final waste volume of 7.6 m^3.

A volume of about 60 m^3 of Low active Liquid waste (corresponding to 15 GBq of activity) has also been produced by various operations before, during and after the operation (flushing of columns, compensation of leakages, level control, emptying of the loop).

3.4.4. Determination of the efficiency of the decontamination operation

The measurement of the gamma activity with ionization chambers placed at several locations allowed to follow continuously the activity level in the primary loop. This is illustrated in figure 4 for the measurement at mid-height of the steam generator.
A map of the gamma radiation dose rate, before and after the decontamination, could be made on the basis of the values measured in the 100 locations selected in the primary and in the purification circuits. The measurements were performed with TLDs and portable teledetectors.

The average DFs obtained are summarized in Table II.

Table II : Overview of the decontamination factors

DF's	Teledetectors		TLD's	
	Contact	Ambient	Contact	Ambient
Primary pipings - primary pumps	9.4	5.6	9.1	3.6
Steam generator	62	33	31	33
Pressurizer	5.2	2.9	3.7	2
Purification System	6.8	3.8	5.5	2.2

As far as the primary loop is concerned, not considering the measurements done in the vicinity of the reactor upper flange where the gamma dose rate is mostly due to the radiations emitted by the activated reactor internals, a mean DF close to 10 has been obtained, with a broad spreading of individual values ranging from 0.1 (redeposition of activity in a horizontal pipe) to 31 (steam generator) according to the measurement location.
As expected, in the vicinity of the reactor upper flange, the impact of the decon process is less important; nevertheless the radiation field at the bottom of the reactor pool has been reduced by a factor around 2.
Along the purification system, where the operating temperature during the decon process was kept lower (40-80°C) than in the primary loop (80-100°C), a mean DF close to 6 was obtained. However, the C-steel ESC condensor treated by the decon solution during the third cycle was not decontaminated at all (DF<1).

In the plant container, the decontamination operation has deeply modified the picture, as far as working conditions are concerned: the ambient dose rate has been reduced by a factor about 10 and amounts now to about 0.08 mSv/h (8 mr/h). In the purification circuit, the ambient dose rate is now around 0.06 mSv/h (6mr/h).

3.5. Radiological aspects of the decontamination operation

The main objective of a decontamination operation is to reduce the collective dose which will be associated to the future dismantling of the loop. Therefore, a very detailed survey of the individual doses associated to the various tasks was performed. The decontamination operation was divided into 3 main phases :

Operation I — A preparatory phase which comprised mainly the closure of the reactor vessel, the repair and maintenance of components and some modifications to the purification system. ***Operation I led to a distributed dose of 135.5 man.mSv***

Operation II — ***The decontamination operation itself led to a distributed dose of only 6.4 man.mSv***

Operation III — ***The post-decontamination operations***, essentially the evacuation of solid and liquid wastes, the opening of the reactor pressure vessel and the evacuation of the resins ***led to a distributed dose of 16.9 man.mSv***

The total dose for the decontamination operation amounted then to 159 man.mSv

The operations required for the closure of the reactor vessel by fastening the reactor cover on the reactor vessel flange are, in a small old-design plant like BR3, still done manually and are therefore responsible for about 70% of the radiation dose associated to the preparatory phase. Nevertheless, the chemical decontamination remains very effective in man-Sv exposure reduction when the dismantling of the primary loop is considered. This is clearly shown in the next Table. The dose savings amounts to about 4.25 man-Sv which means that the collective dose is, thanks to the decontamination, reduced by a factor of 7.5 vs the non-decontamination option.

Table III : Dose savings

	Collective dose man-Sv
Dismantling without DECO	4.9
Decontamination operation	0.16
Dismantling with DECO	0.49
Dose saving	4.25

This evaluation was done at the beginning of the project based on litterature and rough estimate of the dismantling operation. Now that a complete detailed prevision has been made for the dismantling of all the loops of the plant, a more BR3-specific evaluation of the dose savings has been done and amounts to 9 man-Sv saved by the decon operation.

Another important point for the future dismantling operations of the activated and of the contaminated parts is the cleanliness of the surfaces due to the removal of the crud layer. This has facilitated visual inspection of the activated parts and will facilitate the dry cutting of the contaminated parts by avoiding the spreading of loose contamination in the cutting environment.

3.6. Conclusion of the decontamination operation

Some concluding remarks and lessons can be drawn from the experience of a Full System Decontamination operation on a reactor in dismantling phase.

- The process applied is a smooth process, only a few and minor operational problems were encountered during the process. This could only be achieved by a careful and detailed preparation of the operation. It requires a reactor in full satisfactory operational condition and experienced and qualified operators. For a dismantled reactor, if the decontamination strategy is followed, this operation has to be performed maximum 4 to 5 years after the final shutdown.

- The expected dose rate reduction was obtained and man-Sv savings justify the application of a decontamination operation.
- The decontamination operation had an important impact on the dismantling operations of the reactor internals.

  First, a pollution of the reactor pool occurred during the unloading of the reactor internals resulting in a high turbidity and poor visibility in the pool; this pollution was probably due to a small amount of insoluble ferrous oxalate and loose crud still present in the reactor and on the internals. This pollution could easily be removed by the plant filtration system and the water quality could be reset rather easily.

  Secondly, the internals of the reactor pressure vessel were remarkably clean. This allowed to evacuate some activated pieces at the upper part of the reactor as low radioactive waste (dose rate < 2 mSv/h) which would not have been possible without the decontamination. A mass of 2.7 t or 30% of the internals could be evacuated as LLW.

To conclude, the operation was a success from the technological, radiological and economical aspects and such a decontamination operation should be promoted for future dismantling operations in nuclear power reactors.

4. DISMANTLING OF HIGHLY RADIOACTIVE REACTOR INTERNALS

4.1. Description of the reactor internals

The BR3 reactor was first equipped with original Westinghouse internals, from the reactor designer and supplier. In 1964, after 2 cycles, for an experiment called "Vulcain", the internals were exchanged (except the thermal shield) and the old, original ones, were stored in a shielded chamber situated in a corner of the refuelling pool. It was first foreseen to reload the original internals when the Vulcain experiment was achieved, but this was never done, and the Vulcain internals remained in the reactor until the final shutdown. These Vulcain internals have thus undergone the last 9 operating campaigns of the reactor.
As already explained, the whole project covered the dismantling of all the reactor internals : the thermal shield (phase 1), the "Vulcain" internals (phase 2) and the original "Westinghouse" internals (phase 2bis).

4.1.1. The thermal shield and the "Vulcain" internals

Figure 5 gives a general view of the reactor vessel with the "Vulcain" internals. Through a short cylindrical support skirt, the vessel rests on the Neutron Shield Tank, a large annular carbon steel vessel surrounding the reactor vessel and filled with water, which was maintained at low temperature (50°C) during reactor operation. The inner diameter of the reactor vessel is about 1.5 m.
The core (fuel active length : 1 m) is surrounded by the thermal shield, a stainless steel cylinder of 76 mm wall thickness. The thermal shield is the heaviest of the reactor vessel internal pieces (weight : 6 t) and the only one never removed from the reactor vessel during the active plant life.
The other internal pieces can be classified into three sub-assemblies :

1. The lower core support assembly (LCSA) is a stainless steel assembly of about 4.5 m height and 4 t weight. The outer diameter ranges from 1.2 to 1.5 m and the maximum wall thickness is 40 mm. This assembly rests on a cylindrical shoulder at the upper part of the reactor vessel.
2. The upper core support assembly (UCSA) is a stainless steel piece, loaded above the reactor core, maintaining the fuel assemblies in place and, through a spray box, providing part of the safety injection water flow directly above the reactor core. The outer diameter of this piece is about 1 m and its weight is about 300 kg.
3. The collar and its associated instrumentation basket. The reactor vessel collar is a thick ring of carbon steel lined inside with stainless steel (wall thickness : 194 mm; outer diameter : 1715 mm; height : 310 mm). The collar, which was not part of the original design of the reactor vessel, was inserted between the vessel flange and the vessel cover in order to give access to the vessel interior for pipes and instrumentation. Two sets of O-rings and bolts of extended length are used to fasten the collar and the cover on the reactor vessel. The collar is representative (thickness and material) of PWR vessels.

The neutron-induced activity of the internals surrounding the reactor core (LCSA at core level and thermal shield) (see Table IV) is very high. The activation of the reactor collar is very low due to its longer distance from the reactor core. Moreover, there were also guide tubes for the control rods and associated gripping mechanism, as well as the control rods themselves, constituted of borated stainless steel tubes.

Table IV : Specific activity of the dismantled internals

1. Thermal shield (in October 1991, i.e. 4.25 years after shutdown)			
Level vs. midplane (mm)	**Depth (overall thickness = 76mm) (mm)**	**Specific activity (after 4.25 years) (GBq/kg)**	**Specific activity at shut-down, based only on Co-60 (GBq/kg)**
- 238	0 - 19	56	98
	19 - 38	n.d.	
	38 - 57	26.3	46
	57 - 76	19.7	34
+ 248	0 - 19	n.d.	
	14 - 38	40	70
	38 - 57	20.4	36
	57 - 76	19.3	34
+ 735	0 - 19	n.d.	
	19 - 38	5.9	10.3
	38 - 57	1.6	2.8
	57 - 76	3.9	6.8
2. Lower Core Support Assembly (in June 1994, i.e. 7 years after shutdown)			
Level vs. midplane (mm)	**Cut number**	**Specific activity (after 7 years) (GBq/kg)**	**Specific activity at shut-down, based only on Co-60 (GBq/kg)**
- 633	1	3.4	8.5
- 189	2	67.0	168.2
- 213	2bis (core baffle)	125.0	313.9
+ 311	3	19.0	47.7
+ 263	3bis (core baffle)	28.0	70.3
+ 801	4	1.2	3.0
+ 1111	5	0.015	0.038
+ 1918	6	0.0019	0.0048
+ 1626	7	0.00092	0.0023

4.1.2. The Westinghouse internals

The Westinghouse internals are all made in stainless steel (AISI 304). They consist of two main subassemblies : the upper core support assembly and the lower core support assembly. They also include a few other subassemblies. All those subassemblies are described hereafter.

a. Guide tubes hold down plate and ring (see figure 6)
These two pieces are 25.4 mm thick plates. The guide tubes hold down plate is embedded inside the guide tubes hold down ring to form a subassembly of the reactor. The overall diameter is about 1450 mm.

b. Guide tubes (see figure 6)
There are twelve guide tubes. A guide tube has a cylindrical geometry with a conical top end (diameter = 171.5/154; H = 1657 mm). These tubes were used to guide the control rod into the reactor core.

c. Guide tubes support plate (see figure 6)
The guide tubes support plate is a 32 mm thick plate with twelve big holes for the guide tubes.

d. Upper Core Support Assembly (see figure 6)
It comprises two main subassemblies : the upper core support barrel and the upper core support plate.
The upper core support barrel is a cylindrical assembly with top and bottom flanges. The cylinder comprises a circular opening for the water flow to the hot leg. The upper core support plate is a rigid assembly of two perforated plates joined by welding to a spacer ring at their circumference. The overall height of the upper core support plate is 101.6 mm. The upper plate of the assembly supports twelve dashpot stops.

e. Lower Core Support Assembly (see figure 6)
It comprises the following subassemblies :
- the lower core support barrel;
- the reactor core barrel;
- the reactor core baffle;
- the lower core support plate;
- twelve control rod extension shrouds;
- a tie plate at the end of those shroud tubes.

The lower core support barrel consists of a conical section and cylindrical section with top and bottom flanges. The cylindrical section has a nozzle (hot leg) and two diameter guide spacers. It is bolted at its bottom flange to the core barrel and the core baffle.
The core barrel is a cylindrical piece (diameter : 1188/1130 mm; H = 1693 mm) with top and bottom rings. It contains the core baffle (bolted to its upper ring). At its bottom, it is fastened to the lower core support plate by 18 bolts placed top down.
The reactor core baffle consists of a square structure made of plates and ribs with circular top and bottom flanges (see figure 7). Its main thickness is about 6.35 mm.
The lower core support plate is similar to the upper core support plate. The upper part of the lower core support plate supports guide blocks.
The control rod extension shrouds are twelve cylindrical pieces (H = 1286 mm, diameter = 168/154 mm) hanging at the lower core support plate. They are bolted to the lower plate of the core support plate by cap screws. The tie plate is attached by cap screws to the lower end of the shroud tubes.

After 30 years of cooling down, the activity of the internals was still quite high and presented a contact dose rate at the core level much too high to allow direct operation without shielding. The following table gives a summary of the activity of the pieces and a measure of the remaining crud contamination activity still present on the internals.

Table V : Specific activity measurement of the Westinghouse internals

Level vs. midplane	Cut number	Specific activity (after 31 years) (GBq/kg)	Specific activity at unloading based on ^{60}Co (GBq/kg)
-1300	Control rod extension shrouds	0.008	0.47
-403	W1 (Reactor core barrel)	1.805	106.31
-785	W2 (Lower core support plate)	1.206	71.03
-413	W3 (Reactor core baffle)	3.454	203.43
+47	W4 (Reactor core barrel)	1.998	117.68
+37	W5 (Reactor core baffle)	8.114	477.89
+497	W6 (Reactor core barrel)	0.565	33.28
+487	W7 (Reactor core baffle)	2.014	118.62
+997	W8 (Lower core support barrel)	0.009	0.53
+929	W9 (Upper core support plate)	0.073	4.30
+1373	W10 (Lower core support barrel)	0.001	0.06
+1402	W11 (Upper core support barrel)	0.001	0.06

4.2. Dismantling of the thermal shield and comparison of three main cutting techniques (plasma arc torch, EDM, mechanical cutting)

The second part of the phase 1 contract concerned the selection and comparison of cutting techniques for dismantling highly radioactive internals, followed by the actual testing on a first internal : the reactor thermal shield.
The thermal shield is a thick stainless steel cylinder (thickness : 76.2 mm or 3 inches, height : 2.43 m, external diameter : 1.397 m) which surrounds the core and was never unloaded during the whole life of the plant.

Three different cutting techniques were selected for the dismantling of the thermal shield : the plasma arc torch cutting, the electric discharge machining (EDM or sparkling erosion) and the mechanical cutting using a milling cutter (see figure 8).

This allowed to compare three different cutting methods belonging to three types of techniques : a thermal one (the plasma arc torch), an electric one (the EDM) and a mechanical one (the milling cutter). The comparison concerns the amount of generated secondary waste, the cutting duration, the operator's dose uptake and the easiness of the operation.
Other cutting techniques, like laser cutting or high pressure abrasive water jet cutting, were also considered but finally not selected because they were still under development and not yet completely mature for the type of cutting, the environment (under water) and the thickness of the material.

The philosophy was to use existing and proven technologies and to adapt them to the environment and to the application. Great care was paid to the dose forecast (ALARA approach), the secondary waste and the operator's safety.

Considering the secondary waste production foreseen, it was decided to perform plasma cutting in a closed chamber, located in the reactor refuelling pool, allowing to circulate and filter the water as well as the air situated above the water level (filtration of aerosols, evacuation of the produced hydrogen).

The foreseen waste packaging technique implied to cut the thermal shield into pieces having as main dimensions 500 x 540 mm x thickness.
Regarding the geometry of the thermal shield itself, it was necessary to solve the problem of the spacer pins situated at the top level (see figure 9). Indeed, these pins were screwed in place after the loading of the thermal shield in the pressure vessel and then fastened by welding. The pressure vessel presenting diameter restrictions above the thermal shield, it was impossible to unload the complete piece as such, or even a ring of it. Therefore, it was decided to cut vertically the first ring in situ.

The complete cutting scenario is presented on figure 9. It responds to the different constraints given for the operation :

- packaging dimensions < 500 x 540 mm;
- presence of the spacer pins;
- comparison of different cutting techniques;
- no thermal load on the RPV wall;
- plasma torch cutting performed in closed chamber.

After the design and procurement phases, cold testing of the three techniques were carried out on full-scale mock-ups (see figure 10). These cold tests allowed to determine the best cutting parameters, to train the operators and to solve some youth illness of the installations. These cold tests or trials proved to be very efficient in predicting the cutting parameters and in helping to forecast the dose uptake and to optimize the radiation protection.

Moreover, the only part which was not fully tested during the cold tests, for practical reasons, was also the one which gave afterwards the most important problems to solve once in the controlled area. This is one of the *first important results* of this phase of the project : **for important operations or activities related to highly radioactive pieces, the use of cold trials on full-scale mock-ups is really necessary and cannot be avoided.**
It saves finally money and time by allowing to solve problems in an easy environment prior to enter the nuclear environment.

The thermal shield was cut into 40 pieces in 4 months. The pieces were stored at the bottom of the refuelling pool of the plant, awaiting their transfer to the deactivation pool using a specifically designed container.

Table VI hereafter gives the main results of the cutting campaign.

Table VI : Thermal Shield Segmentation - Comparison of the different cutting techniques

Cutting method	Cut length (m)	Kerf width/depth NoxLength (mm)	Cutting speed (1) (mm/min)	Time for cutting + prep. (hours)	Effective cutting speed (mm/min)	Metal removed from thermal shield (dm³)	Secondary waste volume (dm³)	Ratio waste vol/cut length (1) (dm³/m)	Dose uptake (6) (mSv)	Dose uptake per cut length (mSv/m)
Vertic. EDM	2.92	7/76.2 8x365	0.6	184	0.27	1.622	96 (2)	33	7.5	2.56
Horiz. Mech.	4.15	4/18-20 4x4150	6	64	1.08	1.262	26.7 (3)	6.4	8.3	0.66
Horiz. Mech.	4.15	4/18-20 4x4150	6	80	0.88	1.262	26.7 (3)	6.4		
Horiz. Mech.	4.15	4/18-20 4x4150	6	48	1.44	1.262	26.7 (3)	6.4		
Horiz. EDM	4.15	7/76.2 12x346	0.6	240	0.29	2.208	128 (2)	31	7.1	1.71
Vertic. plasma	14.98	11/76.2 24x482 8x426	300	136	1.83	12.55	459 (4)(5)	31	9.2	0.61

(1) For 76.2 mm wall thickness. For EDM, this corresponds to an equivalent speed, the way of cutting being different from the two other methods.
(2) Fine filters containing activated EDM particles.
(3) Coarse filters containing activated chips.
(4) Coarse, intermediate & fine filters; demineralizer columns not completely saturated.
(5) Not taking into account the waste produced by the air filtration system.
(6) The preparatory phase, including the unloading of the reactor internals gave rise to a total dose uptake of 22.6 man-mSv.

The mechanical cutting method produced the smallest secondary waste volume while taking finally less than twice the operating time of the plasma arc torch cutting. Taking into account the long preparation and cold test time as well as the remediation of the plasma dross collecting system problem, the overall operation time is almost similar.
For the dose uptake, the very slow operation speed of the EDM implies a long staying time in the controlled area and in the ambient dose rate (even if low), and then a quite high dose uptake.
The next table gives a summary of the preceding results :

Table VII : Summary of the Thermal Shield dismantling results

Parameter	Cutting speed mm/min (through SS, 76.2 mm thick)		Average effective cutting speed (mm/min)		Dose uptake	Secondary waste volume (for same cut)
Cutting method	**Absolute**	**Relative**	**Absolute**	**Relative**	**Only relative values**	**Only relative values**
EDM	0.6	*1/10*	0.28	*1/4*	*~ 3*	*~ 5*
Mech.	6	*1*	1.13	*1*	*1*	*1*
Plasma	300	*50*	1.83	*1.6*	*~ 1*	*~ 5*

For the produced waste, the values in the table did not take into account the used pool filters of the plant filtration unit. The particles and dust dispersed by the cut in the reactor pool were mostly trapped in these filters.

The distribution of these secondary waste among the different cutting techniques has been made in the following table, using the filter exchange dates and the cutting campaign period as reference.

This is indeed approximative, as some particles can have settled on the bottom of the pool. Nevertheless, it gives also an indication of the pollution of the pool induced by the cutting process.

Table VIII : Use of the pool filtration cartridges from the plant installation
(one filter exchange is constituted by 64 cartridges of 10" long).

	Number of cartridges	Net Waste volume (dm^3)
EDM	3	138.2
Mechanical	1	46.1
Plasma	7	322.6

The results presented above led finally to give the preference to mechanical cutting for the next phases of the project. Indeed, the main advantages of the mechanical cutting can be summarized as follows :

- well known technique, used in workshops; must only be adapted to work under water;
- secondary waste type (chips or swarfs) easily trapped, with a good filling factor of the filter;
- low amount of waste if the tool and the kerf are thin;
- no emission of smoke, gas and dissolved ions;
- overall operation duration comparable to other cutting processes.

4.3. Comparison of different mechanical cutting techniques for dismantling highly radioactive reactor internals

Thanks to the experience gained during phase 1, described above, and taking into account the general geometry of the highly radioactive internals (all internal pieces to be cut have a general shape of revolution of elementary surfaces), it was decided to apply mostly the mechanical cutting technique for their dismantling, where possible.
Two main techniques were selected : the circular saw and the band saw in association with a so-called turntable. The goal was to cut the highly active internals in segments which have a size fitting closely to the final disposal waste package (400 l waste drum).

The main results of the operations using both techniques are summarized hereafter.

4.3.1. Circular sawing

The circular saw (figure 11) was used during phases 2 and 2bis of the project and was foreseen to cut the long cylindrical workpieces horizontally. It was fixed on an extension of the turntable (see figure 12) on which the workpiece was fastened. The circular saw support has 2 degrees of freedom : the X-axis for the feed of the blade into the workpiece (available stroke : about 1 m) and the Y-axis giving the cutting depth (available stroke : about 320 mm). All the movements of the sawing machine as well as the rotational speed of the blade were controlled and monitored remotely.

The following table gives the different blades used for this underwater segmentation.

Table IX : Number of saw blades used

Saw blade		Number of cutter	Rotational speed	Peripheral speed	Cutting depth/ pass	Theoret. max. cutting depth
Diameter (mm)	Pitch (mm)		(Rpm)	(m/mim)	(mm)	(mm)
400	19	3	10	12.6	25 to 30	100
480	20	3	8	12.1	25 to 30	140
610	6	2	7	13.4	25	205
610	19	2	7	13.4	25	205
660	20	2	6	12.4	25	230

The first two showed the best results and straight cuts. But for deep cutting or long distances between the cutting head and the workpiece, the blades with larger diameter (Ø 610 mm or Ø 660 mm) had to be used. The blade with a Ø 610 mm with a pitch of 6 mm was mostly used for cutting thin material (i.e. 1.6 mm thickness) of the baffle.

The following table gives the main results obtained during the cutting operations for the two sets of internals.

Table X : Circular saw cutting results

	Cut section (dm²)	Cutting time during swarfs product. (min)	Total cutting time (min)	Average feed during swarfs product. (dm²/min)	Average production for the whole cut (dm²/min)
Vulcain internals	94.55	6024	23040	0.025	0.004
Westing-house internals	63.24	3295	15006	0.020	0.004
Total/ Mean	157.79	9319	38046	0.023	0.004

The total cutting time includes the operations for changing the clamping pieces, for cutting and for the maintenance of the sawing machine.
During the swarfs production the average feed is about 0.023 dm^2/min. In fact, the feed fluctuates from less than 0.004 dm^2/min to 0.055 dm^2/min following the shape of the workpiece to be cut.

4.3.2. Band sawing

The band saw (figure 13) was foreseen to carry out the vertical cutting of the LCSA and of the reactor vessel collar as well as the segmentation of the plates.

The band saw has a throat of 500 mm and a vertical cutting capability of 960 mm. These dimensions allowed to cut the different pieces to dismantle using the same machine. Thin material (like the core baffle, thickness 1.6 mm) as well as thick annular pieces like the reactor vessel collar (carbon steel clad with stainless steel, overall thickness about 200 mm) can be cut by the machine using adapted saw blades and cutting parameters.
Different blade types (2/3 Teeth per inch, 4/6 Tpi, 6/10 Tpi and 10/14 Tpi) were used depending on the type of cut to be carried out.

Moreover, the machine is able to make horizontal as well as vertical cuts (e.g. for plate segmentation), the blade guides being able to rotate around a horizontal axis. For vertical cutting, the feed is achieved by a controlled vertical motion of the band saw frame while, for horizontal cutting, the workpiece itself is rotated, using the turntable.

The following table gives the main results obtained during the cutting campaigns.

Table XI : Band saw cutting results

	Cutting section (dm^2)	**Cutting time during swarfs product. (min)**	**Total cuting time (min)**	**Average feed during swarfs product. (dm^2/min)**	**Average production for the whole cut (dm^2/min)**
Vulcain internals	216.65	5269	52320	0.041	0.004
Westinghouse internals	160.60	4112	21546	0.039	0.007
Total/Mean	368.14	9381	73866	0.040	0.005

The total cutting time includes the operations for changing the clamping pieces, for cutting and for the maintenance of the sawing machine.

The feed during swarfs production varies quite sensibly, from 0.007 (thin thickness : 1.65 mm) to 0.105 dm^2/min (thick annular piece, overall thickness : about 200 mm).

4.3.3. Swarfs collection

It was originally planned to collect the swarfs during the cut by means of a suction frame surrounding the saw blade. Swarfs were also collected in a funnel with a collecting basket placed under and inside the workpiece. At the end of the horizontal cut of the Vulcain internals, due to frequent blocking of the suction system, the swarfs were not collected anymore during the cut, but were pushed into the funnel, by a water jet after each cut. The remaining swarfs located on the turntable were then sucked off at the end of each cutting campaign by using a straight suction hose.

The total calculated weight of produced swarfs for the whole cutting campaign was 133 kg from which 104 kg were collected by the two methods described above. The remaining 29 kg were located at the bottom of the pool and in the reactor pressure vessel and were removed by suction afterwards.

4.3.4. Summary and conclusion

The next table gives a summary of the two cutting campaigns on the "Vulcain" and "Westinghouse" internals. This gives an overview of the man-power and the dose uptake for the performance.

Table XII : Summary of the man-power and dose uptake during the present cutting campaign

	Cutting length (m)	Cut section (dm^2)	Man-power (man-hours)	Dose uptake (man-mSv)	Working time (hour)
"Vulcain" internals	104.12	311.2	3815	26.98	1256
"Westinghouse" internals	89.02	223.84	2278	9.25	609
Total	193.14	535.04	6093	36.23	1865

The next table gives the qualitative differences and performances of both techniques as they were applied in BR3 for this project.

Table XIII : Qualitative difference between the band saw and the circular saw

	BR3 Circular saw	BR3 Band saw
Cutting force	~ 7 500 N.m.	~ 800 N
Overall volume of the machine	small	large
Horizontal cutting position (level)	only one	several
Cut type	horizontal	horizontal & vertical
Shape of the workpiece	cylindrical or linear without too complex cross section	free with complex cross section
Kerf width	6 mm	2 mm
Depth of passes	25 mm depth per pass	only one pass limited by the opening of the saw blade(500 mm)
Maximum actually cut thickness	192 mm	365 mm
Maximum possible cut thickness	~ 230 mm (for Ø 660 mm)	~ 500 mm (for the BR3 band saw)
Feed for the same cut cross section in SS Thickness : 25 mm 190 mm	5.8 cm^2/min 3.05 cm^2/min	6.02 cm^2/min(*) 7.9 cm^2/min
Swarfs dimensions	great	small
Maximum height of the workpiece	independant	limited (here 900 mm) (by the return way of the blade)
Free space around the workpiece	without requirements inside or behind the workpiece, only limited by the stroke of the saw (320 mm)	both sides of the workpiece free with a minimum of ⌀ 700 mm free behind the workpiece for the wheels

*Limited by the maximum linear feed rate of the machine (25 mm/min).

Both techniques showed to be reliable, usable and efficient.
In the BR3 project, both techniques were complementary. It is obvious that the circular saw technique produces more volume of secondary waste (metal swarfs) due to a greater kerf width. The removed volume of metal (swarfs) is three times higher than with the band saw.
The average cut production (overall cutting speed) for the whole cut was 1.25 times higher with the band saw. Thus, where it is possible (depending on the height, the shape and the existing access on both sides of the piece), it is better to work with the band saw.

In other words, during the general design of a cutting campaign of reactor internals, it is important to minimize the circular saw work and to maximize the band saw within the limits of their capacities, but both remain complementary.

4.4. Other techniques used for the dismantling of the reactor internals

The dismantling of the reactor internals was based on some main techniques and machines as described above. A lot of auxiliary techniques were needed to carry out some specific tasks. These techniques were used to prepare the internals before cutting, to execute some dismounting, to complete a cut begun with a main technique or as a back-up technique. These auxiliary techniques are presented hereafter with the main results obtained and the lessons drawn.

4.4.1. Hydraulic shears

The hydraulic shears allow to cut, very fast and without any production of chips, pieces presenting a relatively small cross section. Remote work at a distance of up to 7 m was possible, the shears being fixed at the end of a long handling tool. If direct vision is not possible, the positioning can be checked using an underwater television system. The cut capacity reached with the hydraulic shears was the following : in full metal about 30 x 7 mm; for tubes OD/ID = 42/35 mm. The replacement of the cutters was needed when the cut edge was not sharp enough anymore. The frequency of replacement depends on the type of cuts which are carried out and cannot be planned easily. Some sets of spare blades are always needed.

The hydraulic shears were used on almost all the BR3 internals (see figure 14):
- on the Vulcain LCSA to dismantle the lower end of the internal and to cut auxiliary tubes (see figure 5);
- on the Vulcain UCSA box to cut the six columns, joining the upper part of the internal to the lower part;
- on the Vulcain instrumentation basket which was almost made of plates and tubes (see figure 15).

The hydraulic shears were also used for two other applications. First, with the band saw machine, to cut the blade and then free the machine when a blade was blocked in the kerf. After placement of a new blade, a new cut could be started just near the first one. The second application concerns mostly manipulation of pieces up to about 20 kg. The shears can be used as a gripper holding pieces which are dismounted.
The negative feature of the hydraulic shears is the place required around the piece to cut (about 10 cm) for the opening of the jaws. Moreover the body of the used machine is about 60 cm long and its diameter about 15 cm.

4.4.2. Core drilling

The core drilling machine (see figure 16) is a mechanical cutting technique which allows to "drill" holes of diameters up to 50 mm. The one used is pneumatically driven and the feed motion is given manually with a long handling rod. The secondary waste produced is composed of chips and a cylindrical core which is extracted from the annular cutter.
The use of this tool was required to avoid problems on the band saw machine during horizontal cuts. Indeed, the saw blade can deviate from its perfect horizontal way, the blade then becoming overloaded and can finally block and/or break. If the saw blade meets at some place a hole, it can go back to its original cut level and the cut can be continued without problems.

On the Upper Core Support Assembly of the Westinghouse internals, 7 holes were drilled (depth : 19 mm). Two core drilling cutters were needed and the whole operation (including the positioning of the machine, the rotation of the piece to present it correctly in front of the cutter, the cutter exchange, ...) took 6 hours, required 27.4 man.hours and gave 0.119 man-mSv.

4.4.3. Unbolter

Unbolting is a very useful technique because dismounting allows to avoid a cut or allows to separate an assembly into subassemblies which have a simpler geometry and are therefore much easier to dismantle afterwards.
When the bolts are placed vertically and the accessibility to their top is possible from a footbridge, the only remaining problem concerns the bolt safety system used. The type of safety system mostly used at BR3, consisting of bended shroud around the bolt head, allowed to place standard sockets on the bolt top. The impact unbolter allowed to break the safety, by the shocks produced by the machine.
For the dismounting of the Rod Shroud Support Plate of the Vulcain internals (see figure 5), standard nuts (M18) were secured by a transversal safety pin (diam.4mm). The unbolting was carried out using reduced distance between the operators and the pieces (2 m water), heavy connecting rod and socket to transmit the power and a heavy pneumatic unbolter. Each safety pin broke after a sequence of bolt-unbolt motions.

When the bolts are placed top down and if the bolts heads are accessible, a hydraulic unbolter handled at the extremity of a long handling tool can be used. The positioning operations have to be made with remote underwater camera viewing.

4.4.4. Reciprocating saw

This saw is based on a mechanical principle. The tool is pneumatically driven and has been adapted to work underwater. The "forward and back" motion of the saw blade represents the cut movement. The feed has to be given by specific supporting device. That tool presents a lot of important advantages :

- The saw blade is fastened only at one extremity. So the accessibility at both sides of the piece to cut is not needed.
- The saw blade length can be up to 600 mm. So it is possible to go very deep into the piece to perform a cut.
- The saw blade has a thickness of about 2 mm. So the kerf is thin and the sawing does not produce too much secondary waste (only small chips).

This technique was used to cut all the tubes penetrating the instrumentation collar (see figure 17) of the Vulcain internals, to cut the collector piece of the Vulcain Upper Core Support Assembly and to achieve horizontal cuts on the Westinghouse core baffle, where the strip to be cut was too deep to be reached by the circular saw.

This tool was also mounted on a frame to cut high activated tubes such as the moderator tubes, rod shroud tubes, control rod... (see figure 18).

4.4.5. EDM

EDM stands for Electro Discharge Machining. This technique is used in workshop to print the shape of an electrode into metallic material. It is also possible to cut metal by using thin plate electrodes. During the comparison of cutting techniques on the thermal shield (see chapter 4.2.1), we showed that EDM was not the best solution to cut thick material. It produces too much secondary waste and the process is too slow.
Nevertheless, for some specific applications, EDM can help dismantling. The biggest advantage of EDM is that almost no force is applied between the electrode and the workpiece and that almost any shape can be given to the electrode. In the BR3 project, the EDM was also used to dismount a plate from equipments which were bolted to it underneath. The bolts were not accessible by standard dismounting tool and we decided to cut the bolts by perforating the plate. Unfortunately, some bolts were placed just under reinforcement ribs of the plate. An oblique perforation EDM system was developed (see figure 19), tested and succesfully applied. The EDM part of the task concerned 38 bolts. Twenty were EDM-perforated vertically and eighteen oblique. Cutting a bolt vertically took in average 0.39 working day (one shift/day) as total operational time and resulted in an average dose uptake of 0.058 man-mSv. Cutting a bolt in an oblique way took in average 0.8 working day and resulted in an average dose uptake of 0.121 man-mSv.
The use of EDM is not recommended as a cutting technique for dismantling thick reactor internals, but, for some "surgery" operations, its flexibility can be a definite advantage. Any surgery EDM work needs a lot of developments and tests. For the dismounting mentioned hereabove, we developed specific electrodes and the positioning system of the EDM-head had to be very precise.

4.5. The dismantling after 30 years cooling down period

One of the objectives of phase 2bis of the project was the comparison of the direct dismantling of internals with the deferred dismantling, after 30 years of cooling down period.
Before starting the dismantling of the cooled down internals (so-called "Westinghouse" internals), measurements of the radiation dose rate were made as well as a theoretical calculation of the dose rate and activity level when unloading.
This theoretical approach was intended to determine the specific activity when unloading to be able to compare the two operations.

The theoretical approach, based on neutronic calculations and activation computations gave quite surprising and interesting results. Indeed, they showed that the build up of the ^{60}Co saturates quite rapidly, after only three or four working campaigns, and tends even to decrease after the first four working campaigns, depending on the initial cobalt concentration in the metal.

This is mostly due to the decrease in ^{59}Co content due to neutron absorption, to the natural decay of the produced ^{60}Co but also to the disappearing of the generated ^{60}Co by radiative capture under neutron irradiation. The combination of these factors give an evolution of the ^{60}Co activity which follows the graph given on figure 20, and explains the quite high level of radiation still present on the "Westinghouse" internals.

When calculating the activity level of the internals at the time of unloading, based on the activity measurement done on the swarfs of the internals and on the decay law of the ^{60}Co, one gets the following table (table XIV).
In this table, the specific activity of the Vulcain internals, being irradiated for 21 years, is given for comparison. One can see that the ^{60}Co specific activity when unloading was indeed of the same order of magnitude. Care must be paid to the initial concentration of cobalt in the steel. This data is not well known for BR3 and can have an important influence (see figure 20).

Table XIV : Comparison of the specific activity of the two internals (in GBq/kg)

Level vs midplane	Westinghouse internals 2 operating campaigns		Vulcain internals 9 operating campaigns	
	in 1995 after 31 years cooling down	in 1964 when unloading	in 1994 after 7 years cooling down	in 1987 when unloading
-413 (baffle)	3.454	203.43		
-213 (baffle)			125.0	313.9
+311 (barrel)			19.0	47.7
+497 (barrel)	0.565	33.28		
+801			1.2	3.0
+929	0.073	4.30		

Nevertheless, the important feature for the comparison of the 2 dismantling operations was that the Westinghouse internals showed representative and similar activity contents when unloading to be able to compare the 30 years deferred dismantling and the almost direct dismantling on the same base.
After a cooling down period of 30 years, the BR3 Westinghouse biggest internal, the Lower Core Support Assembly (LCSA) presented in water at its outside surface at the midplane a dose rate of about 1.7 Sv/h. Inside the LCSA the dose rate rose up to 7 Sv/h. With such a high dose rate level, hands-on dismantling or dry cutting without important shielding is impossible. Therefore, underwater dismantling was the selected method.

For the dismantling of the Westinghouse LCSA, the configuration of the pool where the cuts were carried out, was precisely the same as the configuration used for the immediate dismantling of the Vulcain internals (about 7 m of water). The reactor core mid plane was during the whole cutting campaign under a minimum of 4 m water (even more). Considering that the "half depth" of water is about 12 cm for ^{60}Co, the gamma radiation reduction factor is about 10.000.000.000. The activity reduction factor due to the 30 years cooling down, which is about 50, is negligible compared to this last factor. No significant influence of the cooling down period could then be expected for the direct radiation uptake.

Another important point was the detection of trapped swarfs. During tools and equipment maintenance (out of the water), the detection of high dose rate swarfs is easier than low dose rate swarfs. The high dose rate swarfs, easily detected and located, are then directly eliminated by cleaning while the low dose rate swarfs can sometimes only increase a bit the ambient dose rate. This paradoxical effect can lead to higher dose uptake when working on lower active pieces than on high active pieces.

When comparing the dose uptake for the two sets of internals (see table XI hereafter) the values are higher for the Vulcain internals than for the Westinghouse internals (87 $\mu Sv/dm^2$ and 40 $\mu Sv/dm^2$). This difference is not due to the internals specific activity (shielding by water) but to the change in the ambient dose rate on the reactor operator deck.
Indeed, between phase 2 and phase 2bis, different radiation sources present in the operation area were removed. This removal led to decrease the ambient dose rate from 10-15 $\mu Sv/h$ during phase 2 to 4.5 $\mu Sv/h$ during phase 2bis. This difference explains completely the decrease in the dose uptake experienced for phase 2bis, which is not related to the workpieces activity.

Regarding the waste production, no significant advantage of the cooling down period could be observed. If we compare the immediate dismantling of the Vulcain LCSA and the dismantling of the Westinghouse LCSA (after 30 years), the waste category transition levels are almost the same. The only difference concerned one ring : on the Vulcain internal, the transition from LLW to MLW occured around the upper part of the hot leg; on the Westinghouse LCSA, it occured under the hot leg. So the gain we made on the Westinghouse internals was the category change of 283 kg of material from MLW to LLW. Concerning the production of swarfs, no gain occurred because no specific (complicated) selection was made during the swarfs recuperation.

The main conclusion is thus that no significant advantage can be drawn from waiting for 30 years before dismantling highly radioactive pieces. This seems to be valid for the radioprotection point of view as well as for the waste production.
Regarding the similar activity of both sets of internals at the time of definitive unloading, this conclusion is based on sound comparison results.

4.6. Summary and comparison of the cutting processes

The results produced during the whole cutting campaign of the BR3 internals are summarized in the next table.

Table XV : Summary of the results of the reactor internals dismantling

Technique used	Cut length (m)	Cut section (dm^2)	Time for cutting & preparation (hours)	Effective cutting speed* (dm^2/hr)	Dose uptake (man-mSv)	Dose/dm^2 (mSv/dm^2)
Phase 1 : Thermal Shield						
EDM	7.1	54	424	0.13	14.6	0.271
Circ. saw	12.5	95	192	0.49	8.3	0.087
Plasma	15.0	114	136	0.84	9.2	0.081
Total	**34.5**	**263**	**752**	**0.35**	**32.1**	**0.122**
Phase 2 : Vulcain Internals						
Circ. saw	43.4	95	384	0.25	9.1	0.096
Band saw	60.7	217	872	0.25	17.9	0.083
Total	**104.1**	**311**	**1256**	**0.25**	**27.0**	**0.087**
Phase 2bis : Westinghouse Internals						
Circ. saw	30.2	63	250	0.25	3.8	0.060
Band saw	58.8	161	359	0.45	5.3	0.033
Total	**89.0**	**224**	**609**	**0.37**	**9.1**	**0.040**
Whole operation						
Total	**227.6**	**797.9**	**2617.2**	**0.30**	**68.1**	**0.085**

* Effective cutting speed : it is the time between two cuts (installation, cut, cut piece removal and clamping adaptation).

It is important to remember that the work piece of phase 1 was the thermal shield (a quite simple cylinder of 1.5 m diameter and 3 inches thickness, or 76.2 mm). For phases 2 and 2bis the workpieces had a more complex geometry with different plate thicknesses, different diameters, protrusions, etc.

The next table (table XVI) gives an overview of the secondary waste volume produced during the job. The waste volume is the sum of :
- the volume of the coarse filters from the local filtration unit installed in the reactor pool to collect the swarfs directly at the source;
- the volume of the fine filters from the local filtation unit installed in the reactor pool;

- the volume of the fine filters from the filtration unit of the plant;
- and, finally, the worn tools of the different equipments (saw blades, electrodes, ...).

Table XVI : Overview of the secondary waste volume

Technique used	Cut section (dm²)	Amount of metal released (dm³)	Secondary waste volume (dm³)	Ratio waste vol/ cut section (dm³/dm²)
Phase 1				
EDM	53.87	3.77	377.2	7.0
Circular saw	94.87	3.79	130.3	1.4
Plasma	114.15	12.56	781.6	6.8
Total	**262.89**	**20**	**1289.1**	**4.9**
Phase 2				
Circular saw	94.55	5.67	199.5	2.1
Band saw	216.65	4.33	292.7	1.4
Total	**311.20**	**10.01**	**492.2**	**1.6**
Phase 2bis				
Circular saw	63.24	3.79	111.2	1.8
Band saw	160.60	3.21	105.5	0.7
Total	**223.84**	**7.01**	**216.7**	**1.0**
Pool cleaning				
Pool cleaning	-	-	153	-
Whole operation				
Total	**797.93**	**37.02**	**2151.0**	**2.7**

If we compare the results obtained with the mechanical technique, it is important to note that the thickness of the circular saw blade or of the band saw blade varied during the different phases : 4 mm (phase 1), 6 mm for the circular saw blade and 2 mm for the band saw blade (phase 2 and 2bis).
Moreover, during phase 2, the fine filters of the local filtration unit were not used during a part of the cutting campaign. The total volume of these filters is about 116 dm³. Thus, the results obtained during phase 2 are of the same order of magnitude than those of phase 1 and the results of phase 2bis are better than the others (better filling of the coarse filters).
At the end of the whole project a deep clean up of the reactor pool bottom has been carried out.

It resulted in the production of 9 supplementary strainers (about 153 dm³). These waste can not be related to any one of the project phases but has to be added to the overall secondary waste volume production. Another lesson learned during the two last phases is the improvement of the prediction of the necessary worktime. Table XVII illustrates this.

Table XVII : Comparison of observed and predicted manpower and dose uptake

Technique		Phase 2			Phase 2bis		
		Predicted	Observed	Ratio observed/ predicted	Predicted	Observed	Ratio observed/ predicted
CIRCULAR SAW	Cut section (dm²)		94.55			63.24	
	Time for cutting (hrs)	312	384	1.23	196	250	1.28
	Manpower (man-hours)	930	1444	1.55	804	930	1.16
	Dose uptake (man-mSv)	13.85	9.05	0.65	6.01	3.77	0.63
	Dose uptake/dm² (mSv/dm²)		0.096			0.060	
BAND SAW	Cut section (dm²)		216.65			160.60	
	Time for cutting (hrs)	504	872	1.73	426	359	0.84
	Manpower (man-hours)	1443	2524	1.75	1725	1347	0.78
	Dose uptake (man-mSv)	21.65	17.94	0.83	8.68	5.28	0.61
	Dose uptake/dm² (mSv/dm²)		0.083			0.033	
TOTAL	Cut section (dm²)		311.20			223.84	
	Time for cutting (hrs)	816	1256	1.54	622	609	0.98
	Manpower (man-hours)	2373	3968	1.67	2529	2277	0.90
	Dose uptake (man-mSv)	35.50	26.99	0.76	14.69	9.05	0.62
	Dose uptake/dm² (mSv/dm²)		0.087			0.040	

During *phase 2* the total observed time in hours and man-hours is about 50% higher than predicted. This difference comes from the operations that were not taken into account due to the lack of experience for all problems occurring during the work.

Afterwards this was taken into account and estimated to be about 30% of the theoretical cutting time. For the manpower, the difference is due to the same reason but also to the continuous presence of partners delegates during the whole phase. You can note the divergence between the results of the circular saw and the band saw. In fact, during phase 2, a problem occurred with the band saw and it was stopped during 13 whole days on a total of about 109 working days (12% of the working time).

During *phase 2bis*, the total predicted and observed hours and manpower are in good accordance. For this prevision several factors were taken into account :
- a supplement of 30% for all problems occurring during the work;
- the experience gained during phase 2;
- some pieces were dismounted and not cut.

During this phase, a difference in working time and man-hours appeared between the two machines. Here, due to a problem with the circular saw, the machine was stopped during 79 hours on a total working time of 250 hours (32% of the working time).

Although the optimisation of the cutting parameters is important to have a reliable technique, it is important to try to reduce the times allocated to the other operations (installation, saw blades exchange, cut pieces evacuation, filtration devices and filter exchange, etc.).

Comparison of two dismantling methods

The comparison between the two sets of internals shows that there are two similar cuts situated at the level of the bolted link between the core support plate and the core barrel. The dismounting methods are different for both internals. The circular saw has been chosen to cut the Vulcain internals, while an unbolting device to dismount the Westinghouse internals seemed the most appropriate. This difference in the dismantling approach comes from the fact that for the Vulcain internals, some pipes were present through the core support plate and the dowels were located very far from the core barrel. Therefore, it was necessary to use the circular saw to cut it.

To balance the different results obtained, it is interesting to compare this one with a normal cut result (without any bolted links).

The next table gives the comparison.

Table XVIII : Comparison of two dismantling methods for bolted assemblies

	Type of operation	Total cutting time (hours)	Manpower (man-hours)	Dose uptake (man-mSv)	Dose rate equivalent (mSv/h)*
Phase 2	Circular saw	108	408	1.63	0.004
Phase 2bis	Manual unbolting	26	94	0.37	0.004
Reference : Phase 2bis : Standard cut	Circular saw	36	135	0.73	0.005

*Note : The dose rate equivalent is a calculated mean dose rate, based on the actual man-hours and mSv uptake.

Where it is possible, unbolting is the best solution. The total operation time is shorter. Direct access and direct vision facilitate the work. Compared to a standard cut without bolted links, the two operations gave similar results. In addition, the use of a standard machine reduced the operational total cost.

5. THE ALARA APPROACH AND THE DISTRIBUTED DOSE

Due to the low number of dismantling projects already performed and because dismantling happens only once for each specific installation, there was an important lack of knowledge about the technical feasibility and about the radiological costs of a whole representative dismantling project. Therefore, the radioprotection has been extensively analysed and applied throughout the project. This topic is a very important one in the scope of a global analysis and development programme on dismantling of large nuclear installations.

The previous chapters described the technical part of carrying out such operations (decontamination, remote segmentation, dismantling of high active internals). As far as radiological optimization is concerned, the ALARA approach during the dismantling operations has been implemented and improved during all the operations presented in this final report.

The ALARA principle is the basic idea of the optimization of radiation protection and is based on 3 fundamental principles (ICRP - Publication 26) :

1) no practice shall be adopted unless its introduction produces a positive net benefit (justification principle);
2) all exposures shall be kept *as low as reasonably achievable*, economic and social factors being taken into account (optimization principle);
3) the dose equivalent to individuals shall not exceed the limits recommended and legal (limitation principle).

Thanks to the elaboration of a general ALARA programme, we have tried to implement this principle at all stages of dismantling works performed at BR3 till now.

5.1. SCK•CEN experience

During the Phase 1 (Decontamination of the Primary loop - Realisation of cutting equipment - segmentation of a first reactor internal), SCK•CEN took a sub-contractor, the French company CEPN, which is specialized in radioprotection optimization, to provide assistance concerning the radiological optimisation of the decontamination of the BR3 primary loop. The collaboration with CEPN provided the theoretical support, guidelines and key elements for the implementation of the ALARA programme during the decontamination. It consisted in making an initial detailed subdivision of the work procedure, in defining the work area and the related average dose rate as well as the required manpower needed for each task. This detailed analysis allowed the tasks giving the main contributions to the total dose to be identified and also the possibilities for reducing the radiation exposure at these critical points to be evaluated. Finally, a comparison between predicted and received doses has been made a posteriori in order to improve the prediction model.

It was intended to conduct the same ALARA analyses for other tasks of the pilot dismantling project.

As has clearly been shown during the collaboration with CEPN, the implementation of an ALARA programme corresponds to the application of a traditional work management approach in the field of radiation protection, namely :

- prediction of doses during the preparation phase of the operation,
- performance follow-up and corrective actions during the operation itself,
- feedback analysis after the operation.

More in details, the general philosophy of the ALARA approach, based on the above principle, presents many radiological optimization aspects that are dealt with during dismantling operations :

1. The so-called "Predictive ALARA Plan"
   1.1. Choice of the technical process to carry out the work : different processes can be chosen, only the best must be applied and optimized. The optimization is carried out adapting the tools and the work procedure.
   1.2. Determination of elementary tasks :
   Obviously, for each dismantling operation, it is mandatory to define several elementary tasks that will be quantitatively evaluated. This evaluation of the elementary tasks consists in a review of all the different options and possibilities that can be applied : need and use of shielding, technical options, choices of tools and equipments.
   1.3. Prediction of doses related to those tasks : the manpower and the workload to execute each task has also to be evaluated. Some analytical tools may be needed to assess the dose rate during a specific operation.
   1.4. Definition of the dosimetric target corresponding to the Alara principle in order to decide the acceptance of the corresponding working procedure by Radiation Protection Management, and to verify that all individual doses do not exceed legal limits.
   1.5. Application of working methods that are fully feasible according to the ALARA principle, it is to say as they were defined and optimized in the previous points. ALARA reviews and checklist might be used at this stage.
2. Organization of the follow-up of the job-related doses : a structure to monitor tasks on a general and daily basis has to be settled so that reliable records may be obtained on the various parameters to be assessed : working time and dose uptake of workers needed for the specific investigated job (and not diluted on several jobs).
3. Feedback analysis after the operation : the reliable records provided by daily follow-up can be incorporated into some data bases. This will make traceability and reporting easier on a later stage. The comparison between predictions and observations enables to improve the predictive model for next operations. This analysis will give some lessons to be learnt for future operations. A detailed analysis of the operation may point out what type of problems are occurring, and so, to prevent or reduce the other occurrences of this kind of problems.

It is obvious that such an ALARA approach requires the commitment of the management team of the project which must establish an appropriate ALARA structure (Committee, project group, coordinator) and support its organisation by providing it with tools and on time information. This structure has been set up progressively during the BR3 decommissioning project.

Another important aspect not really described here, is the motivation of the workforce that has to be aware of the benefit of such a working procedure.

As explained before and in previous progress reports, the dismantling of BR3 has been conceived at first, as a succession of technical phases aiming at, first, decontaminating, then, testing several dismantling techniques, finally applying the tested techniques during the dismantling of high activated internals of BR3 reactor.

In practice, to apply this ALARA approach for dismantling the BR3, the philosophy has been the following:

A.1. To decontaminate the primary loop to reduce the ambient dose rate surrounding the working areas.

B.1. To carry out shop tests with small-size cutting prototypes.

B.2. To define specifications for the cutting equipments : remote control equipment, smooth surface roughness for easy decontamination, easy maintenance, high MTBF (Mean Time Between Failure), etc.

B.3. To test the specially developed tools on full-size mock-ups at BR3 in unrestricted area to train the personnel with the machine in conditions similar to the ones they will face in controlled area. This step assesses the time needed for similar jobs in controlled areas and represents a main parameter for the dose uptake prediction, as well as for the dose reduction by training of the personnel and solving readily encountered problems.

B.4. To prepare the work area in the reactor building (arrangement of the working area : decontamination of the refuelling pool, filling the pool with 6.9 m water above the reactor upper flange, positioning of other shielding on the footbridge, ...).

B.5. To introduce the equipments in the reactor building, verify and position it in its working configuration and to perform the work on site with the required personnel.

B.6. To evacuate the HLW and other waste as soon as possible to put them in safe conditions without implying dose uptake for the operators.

Several options (water levels, use of shielding material in general, need of decontamination, choice of decontamination process) were evaluated on the basis of a cost-benefit analysis for most of the performed operations.
At this moment, it is only possible to estimate the radiological gain due to the decontamination operation. The real radiological justification will be observed during the dismantling of the other parts of the primary circuit (steam generator, pressurizer, auxiliary circuits, ...) where the ambient dose rate level has been reduced significantly.
It is also obvious that working under water for the dismantling of the internals reduces drastically the dose distributed to the workers (only 4 m water above highly active components provide a dose reduction factor of about 10^{10}) and consequently, that most of the improvements can be done on maintenance operations (when the tools are removed out of the water), on the decontamination ability of the equipment and on the operations duration.
The dosimetric impact of the preparation of the decontamination operation was mainly due to the manual closure of the reactor pressure vessel but was also due to the necessity to perform a deep and complete review and checking of the functionality of all the equipments and instrumentation of the plant with even some repairs and improvements. This again shows the necessity to maintain the circuits in operating conditions and so to perform a decontamination operation not too long after reactor shutdown. Some modifications to the circuit were also necessary to implement the decontamination.

For some cutting operations, for example the EDM technique during the dismantling of the so-called "rod shroud tubes support plate" (RSSP), a learning effect as shown in figure 21 can occur because of the repetitive or symmetrical nature of these operations.
In order to analyse more deeply all data concerning long duration dismantling operations, an accurate follow-up of doses was performed. First, an electronic computerized dosimetric system, has been put into service in March 1992. With this system, indications about residence times at different levels of radiation could be obtained. Later, it was decided to implement the use of an on-site data base at the BR3 site. This data base had to enable several important tasks :

- collecting data from the electronic dosimeter;
- close dose follow-up of performances (individual and collective dose uptake related to specific operations);
- thorough analysis of various parameters (problems, working time);
- making further reporting easier;
- improving the documentation and the traceability of information.

The implementation of a complex data base has been carried out in 2 steps corresponding to 2 previously described cutting operations executed in 1994, i.e. the horizontal cutting of the LCSA and the vertical cutting of the LCSA from the Vulcain internals. The two sources of information used were the logbook in which the engineers and operators wrote all useful information and the computerized dosimetric system recording the time duration and dose intake for each worker.

One improvement in the dose reduction was also the implementation of the use of individual electronic dosimeters with sound alarm. This simple but efficient system prevents operators to stay in areas where the dose rate is higher.

5.2. Distributed Dose

Table XIX shows the distributed dose for the main phases of the pilot project as well as the ratio between predicted and observed dose.

Table XIX : Summary of the predicted and observed dose uptake

	Predicted Dose (man-mSv)	Observed Dose (man-mSv)	Ratio Predicted / Observed
Phase 1 :			
Full-System Decontamination	148.5	158.8	0.94
Segmentation of the Thermal Shield	82.9	55.2	1.50
Phase 2 : Vulcain Internals			
Upper Internals Dismantling	23.7	10.8	2.19
Horizontal Cutting of the LCSA	18.0	11.2	1.61
Vertical Cutting of the LCSA	22.6	20.62	1.10
Phase 2 Bis : Westinghouse Internals			
Horizontal Cutting of the LCSA	6.0	3.8	1.59
Vertical Cutting of the LCSA	8.7	5.5	1.58

The first conclusions drawn from this table is that when the project went on, the dose commitment for each important operation tends to be reduced drastically (from 55.2 man-mSv for the segmentation of the thermal shield to less than 10 man-mSv for the whole dismantling of the Westinghouse LCSA). The predicted dose uptake remains overestimated by a factor close to 1.5, even when the total dose uptake went down, and thus when the dose rates were very low and the systematic errors can normally increase.

Figure 22 shows an example of the dose and manpower follow-up for a long duration dismantling operation, the horizontal cutting of the Vulcain LCSA.

5.3. Conclusion

Although dismantling tasks are quite specific, compared to maintenance and operational tasks, an ALARA programme, as described above, is totally enforceable if such an operation is prepared, followed and analysed from a radiation protection point of view. The ALARA Principle must be incorporated into all levels of decision-making which occur during decommissioning operations.

6. WASTE MANAGEMENT

6.1. Waste minimization

During the whole project, the waste minimization was one of the main objectives of the operating team.
This objective follows two goals, common to all dismantlers :

- to minimize the cost, regarding the very high cost of radwaste conditioning and disposal;
- to reduce, as far as possible, the doses associated with the handling of the radioactive waste; this follows the overall ALARA principle.

Waste minimization was followed from the beginning of the project but the optimization improved as the project went on, regarding the experience gained.

6.1.1. Waste minimization during the Full System Decontamination

The selection of the decon process as well as its execution was carried out in view of minimizing the produced secondary waste as far as possible, while answering to the Belgian Waste agency requirement.
During the selection of the process, the foreseen generated waste volume, as well as the physico-chemical form of the generated waste, played an important role in selecting the process. The CORD process showed *theoretically* to be the one producing the less amount of waste.
During the process itself, the operation was stopped when the dose rate decrease was sufficiently high. A new operation cycle would have added more resins while decreasing further the dose rate only marginally.

6.1.2. Waste minimization during cutting

During the different cutting operations, the waste minimization was also one of the important topics analysed. Phase 1 of the pilot project gave some learnings about the waste generation of different cutting techniques.
One of the main results of this phase concerned the secondary waste produced per cut meter and square meter (see table VI).
It is then evident that not only the kerf width plays an important role in the generation of secondary waste, but also the physical form of the waste (swarfs, dross, sedimented particles, suspended particles, hot particles, particle size distribution), the necessary auxiliary equipments which have to be used to perform the cuts, the collection of the waste and the authorized waste package.

The main conclusions from this operation can be summarized as follows :

- One has to keep the kerf width as thin as possible.
- Fine particles are only trapped on small porosity filters with a very low capacity, so one has to keep the particles or waste overall dimensions as large as possible.
- The particle size distribution of the produced waste should be kept as narrow as possible. A broad spectrum implies the use of different filtration media and then multiplies the quantity of secondary waste.

- Auxiliary material and equipment should be kept free of contamination or easily decontaminated. Complicated geometries, rough surfaces and closed spaces should be completely avoided. This "tertiary" waste, if imperfectly designed can generate large amounts of waste.
- It is not yet clear whether a swarf collection at the source is better, or whether it is better to collect the produced swarfs afterwards. This depends mostly on the specific environment and the technical feasibility of the direct swarfs collection. For small particles collection, the direct collection is surely necessary.

The operations carried out in phase 1 have thus shown different aspects which were taken into account during the following phases of the project.
The same principle applied for the dismantling of the remaining internals, where still other activities were taken to minimize the waste production :
- avoid cuts when not needed (e.g. unbolting fastening bolts instead of cutting them);
- prepare the cutting scheme in view of minimizing the number of cuts taking into account the waste packaging constraints;
  (N.B. The 400 l drum waste package and the limited waste quantity of 700 kg/400 l drum which were imposed by the National Waste Agency, led to low waste packaging filling ratio.)
- use tools producing thin kerfs (e.g. the band saw with a kerf of about 2 mm);
- compare and optimize the waste collection, at the source or after the operation.

6.2. Evacuation of the High Level Waste (HLW)

The quantity of waste, produced during the cutting of the Thermal Shield, the Vulcain and the Westinghouse BR3 internals, amounts to 23.4 tons. From this total, the high or medium level waste represents 12.7 tons (54.2%).
The table below gives the total amount of waste, produced during the three major cutting campaigns and separated in the three waste categories, namely high level waste (HLW; contact dose rate ≥ 0.2 Sv/h), medium level waste (MLW; 0.2 Sv/h > contact dose rate ≥ 2 mSv/h) and low level waste (LLW; contact dose rate < 2 mSv/h).

Table XX : Amount of waste produced during the dismantling of the internals

	HLW (t)	MLW (t)	LLW (t)	Total (t)
Thermal Shield	5.5	0	0	5.5
Vulcain internals	2.8	1.0	7.4	11.2
Westinghouse internals	3.4		3.3	6.7
Total	12.7		10.7	23.4

Remark : The difference between HLW and MLW was not made during the cutting of the Westinghouse internals for reasons of waste volume optimization

The following table gives the quantity of swarfs and the secondary waste (strainers, worn tools, filters, etc...) for the three cutting campaigns.

Table XXI : Distribution of the secondary waste between swarfs and other waste

	Swarfs (dm^3)	Secondary waste (dm^3)
Thermal Shield	20	1289
Vulcain internals	10	492
Westinghouse internals	7	216
Pool cleaning	-	153
Total	37	2150

Most of the swarfs were collected during the campaign in 21 strainers (total waste volume : 357 dm^3). Besides the swarfs, other highly activated pieces (bolts, little pieces of metal, ...) were put in those strainers.
The high and medium level part of the waste, the strainers and the high active filters, were transported to Belgoprocess for conditioning and interim storage. The transports required a shielded container, specially designed for the evacuation of the BR3 HLW and MLW. A total of 40 transports were carried out during three evacuation campaigns.

For one transport, the waste packaging was as follows : the cut pieces are positioned in a dedicated rack. To get a minimum of waste volume, each piece of waste (sometimes with a very complex geometry) had its own particular position in a rack. Two racks form one basket; the basket is loaded into the transport container (see also figure 23). After transport, the basket is unloaded at the waste conditioner facility and placed directly in a 400 l drum. The drum is filled with cement. These operations were performed in a hot cell.

Besides the cut pieces coming from the different reactor internals, other highly activated pieces were put into the baskets for transporting to the waste conditioner. These were mostly produced during the 25 years life of the BR3 reactor. The total weight of those pieces amounted up to around 200 kg.
The next table gives the most important data on the three HLW/MLW evacuation campaigns.

Table XXII : HLW/MLW evacuation campaigns summary

	Number of transports	Weight waste (tons)	Transported activity (Co-60) (TBq)	Workload (man-Hr)	Dose (man-mSv)
Campaign 1	13	5.8	97	311	3.7
Campaign 2	14	3.2	27	354	2.8
Campaign 3	13	3.9	10.5	354	1.7
Total	40	12.9	134.5	1019	7.9

The medium active filters (MLW), packaged in 200 l drums, were transported to the waste conditioner with another shielded container. The low-level waste was packaged in standard 200 l and 400 l drums, ready to be evacuated to the waste conditioner.

7. SUMMARY OF THE MAIN RESULTS

The different parts of the project were described in the preceding chapters, including their detailed results.
As a summary, some concluding results can be pointed out.
First, for the pre-dismantling *decontamination* of the primary loop, this operation has shown its advantages :

- it reduces significantly the overall dose rate around the primary loop and in the work area; the total dose uptake balance shows a significant advantage to perform this operation;
- even for the dismantling of the internals, it allows to change the waste category by removing the hot spots due to the contamination;
- the total amount of radwaste produced by this operation can be limited, but it is difficult to forecast and to control precisely the amount of dissolved ions produced by the operation (non uniformity of the crud deposits).

Moreover, some further lessons can also be drawn. Indeed, it is important to execute the operation quite soon after the plant shutdown to be able to reuse the plant installations and equipment for the operation.

For the *dismantling* of the reactor internals, different cutting techniques were compared. This has shown that, for highly radioactive internals, the mechanical cutting methods can be used easily and gave very good results concerning the produced waste, the contamination of the environment, the cutting duration and the dose uptake.

Different mechanical cutting methods were used and compared, and each of them showed their advantages and drawbacks. For the dismantling of large pieces presenting complicated shapes, the band sawing has proved to be very effective.
Moreover, carrying out the operations under water prevented to produce a high dose uptake to the operators. The presence of two sets of internals, having undergone different cooling down periods (7 years and 31 years), has shown that the difference in activity due to the natural decay of ^{60}Co does not influence significantly the operator's dose uptake. Also, the cooling down period of 31 years of the first set was not sufficient to imply any improvement or any important advantage for dismantling these pieces.

Another interesting result was that after only two years of operation, the first set of internals presented at the time of unloading, an activity of the same order of magnitude as the one having been irradiated for 21 years.

Finally, figure 24 summarizes the main parameters for all the activities of this project. The total manpower includes the design, the cold test and the performance. The dose uptake corresponds obviously only to the performance.

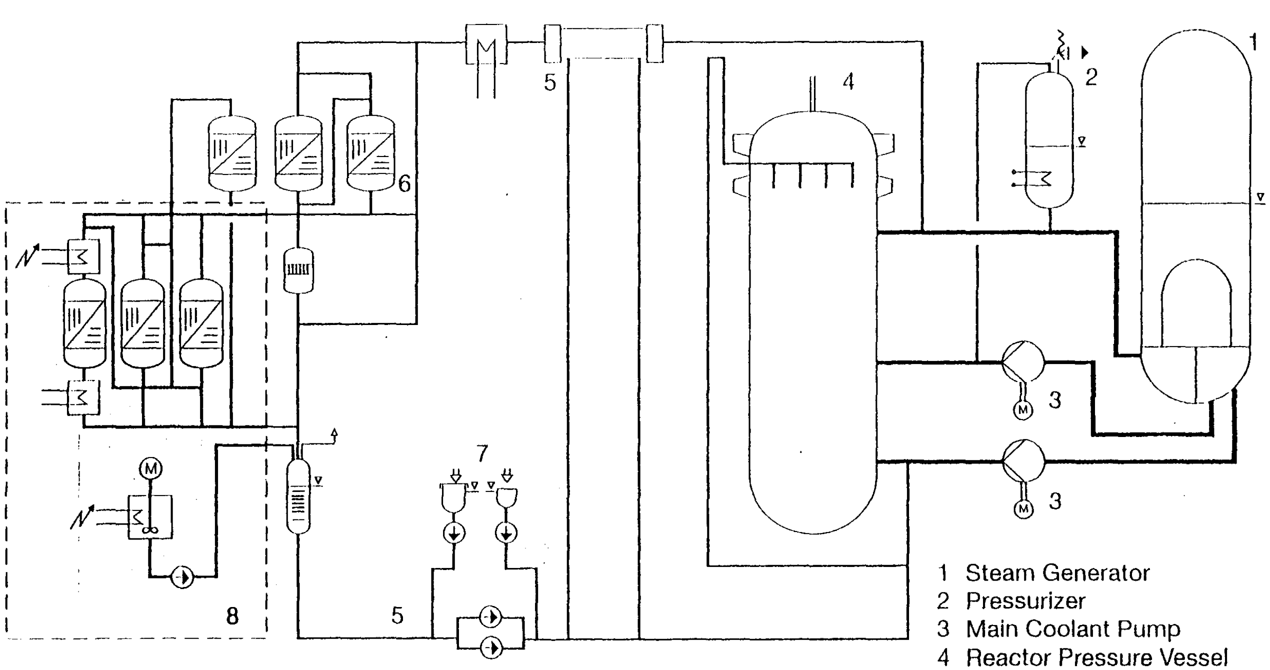
8. CONCLUSIONS

This pilot decommissioning project, supported by the European Commission in the framework of its five year RTD programme on the decommissioning of nuclear installations, has brought interesting results for future decommissioning of nuclear power plants and installations. Moreover, as a pilot project, the BR3 project allowed the different European contractors and partners to build up experience in this field, which is now developing quite fast.

The project has associated industries from different countries in the European Union and got international reputation through its multinational character. Moreover it was the first one dealing with the dismantling of a Pressurized Water Reactor (PWR), which is one of the reactor types the most widely spread in Europe and in the world.

Research and Technical Developments have been carried out throughout the project and allowed to get interesting results developed in the report. Nevertheless some Development works and even Research are still needed to continue improvement of the operations, the procedures, the technologies, etc., in order to reduce again the dose uptake and the waste production. This is probably important for an activity rapidly growing in the nuclear sector, and even for the nuclear industry itself.

FIGURE 1



1 Steam Generator
2 Pressurizer
3 Main Coolant Pump
4 Reactor Pressure Vessel
5 Volume Control System
6 Coolant Purification System
7 Chemical Dosing System
8 AMDA

Decontamination of the Primary Circuit
of the BR 3 with the CORD process

Decontamination

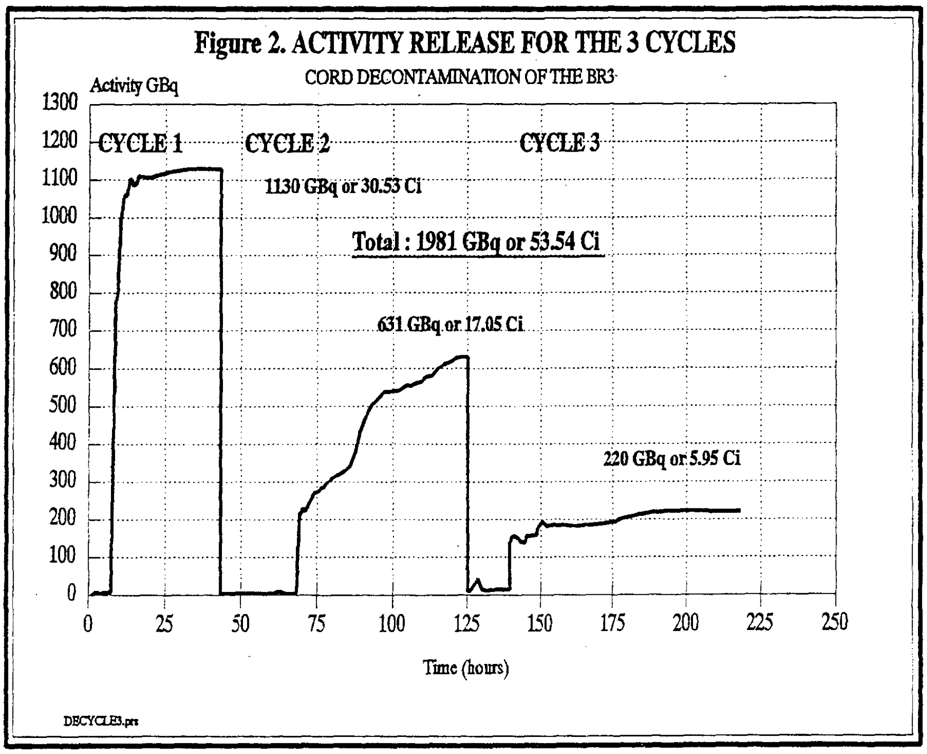


Figure 2. ACTIVITY RELEASE FOR THE 3 CYCLES
CORD DECONTAMINATION OF THE BR3
Activity GBq
CYCLE 1
CYCLE 2
CYCLE 3
1130 GBq or 30.53 Ci
Total : 1981 GBq or 53.54 Ci
631 GBq or 17.05 Ci
220 GBq or 5.95 Ci
Time (hours)
DECYCLE3.prs

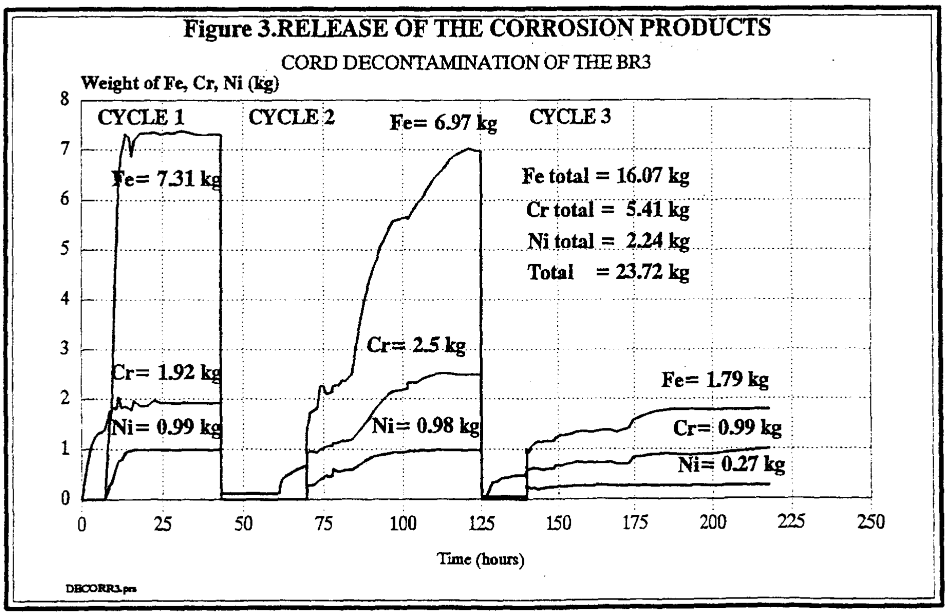
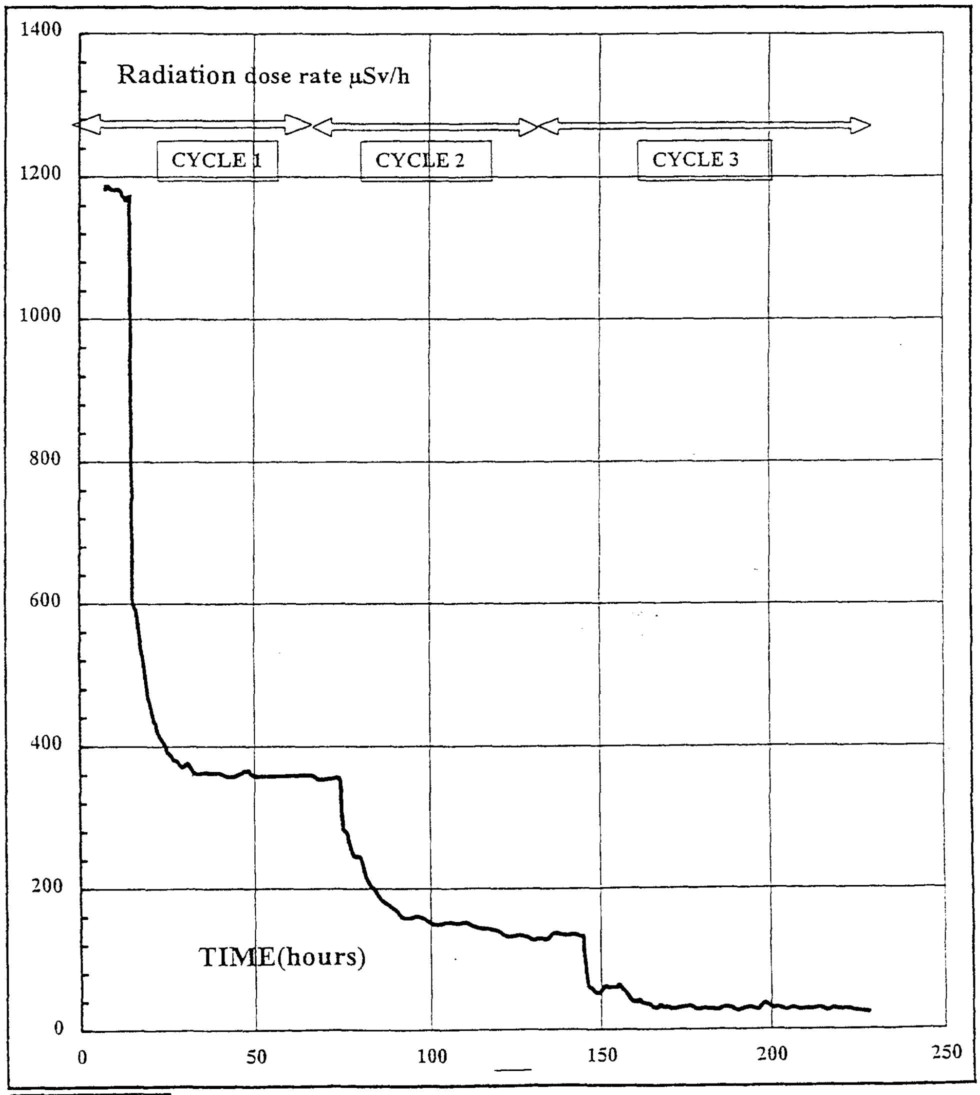


Figure 3.RELEASE OF THE CORROSION PRODUCTS
CORD DECONTAMINATION OF THE BR3
Weight of Fe, Cr, Ni (kg)
CYCLE 1
CYCLE 2
Fe= 6.97 kg
CYCLE 3
Fe= 7.31 kg
Fe total = 16.07 kg
Cr total = 5.41 kg
Ni total = 2.24 kg
Total = 23.72 kg
Cr= 2.5 kg
Cr= 1.92 kg
Fe= 1.79 kg
Ni= 0.99 kg
Ni= 0.98 kg
Cr= 0.99 kg
Ni= 0.27 kg
Time (hours)
DBCORR3.prs

Fig 4. RADIATION LEVEL STEAM GENERATOR
3 DECONTAMINATION CYCLES



RAD-SG3T

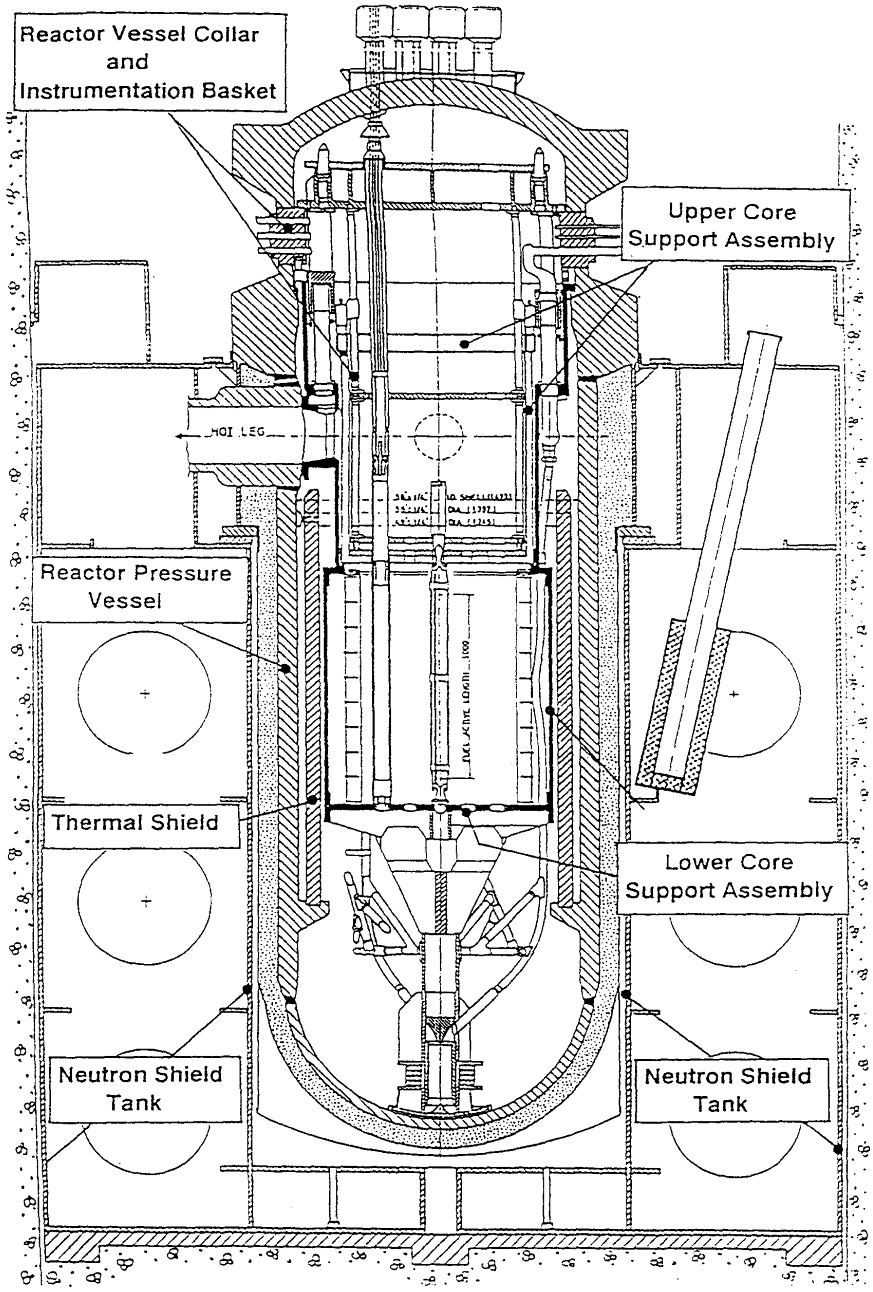
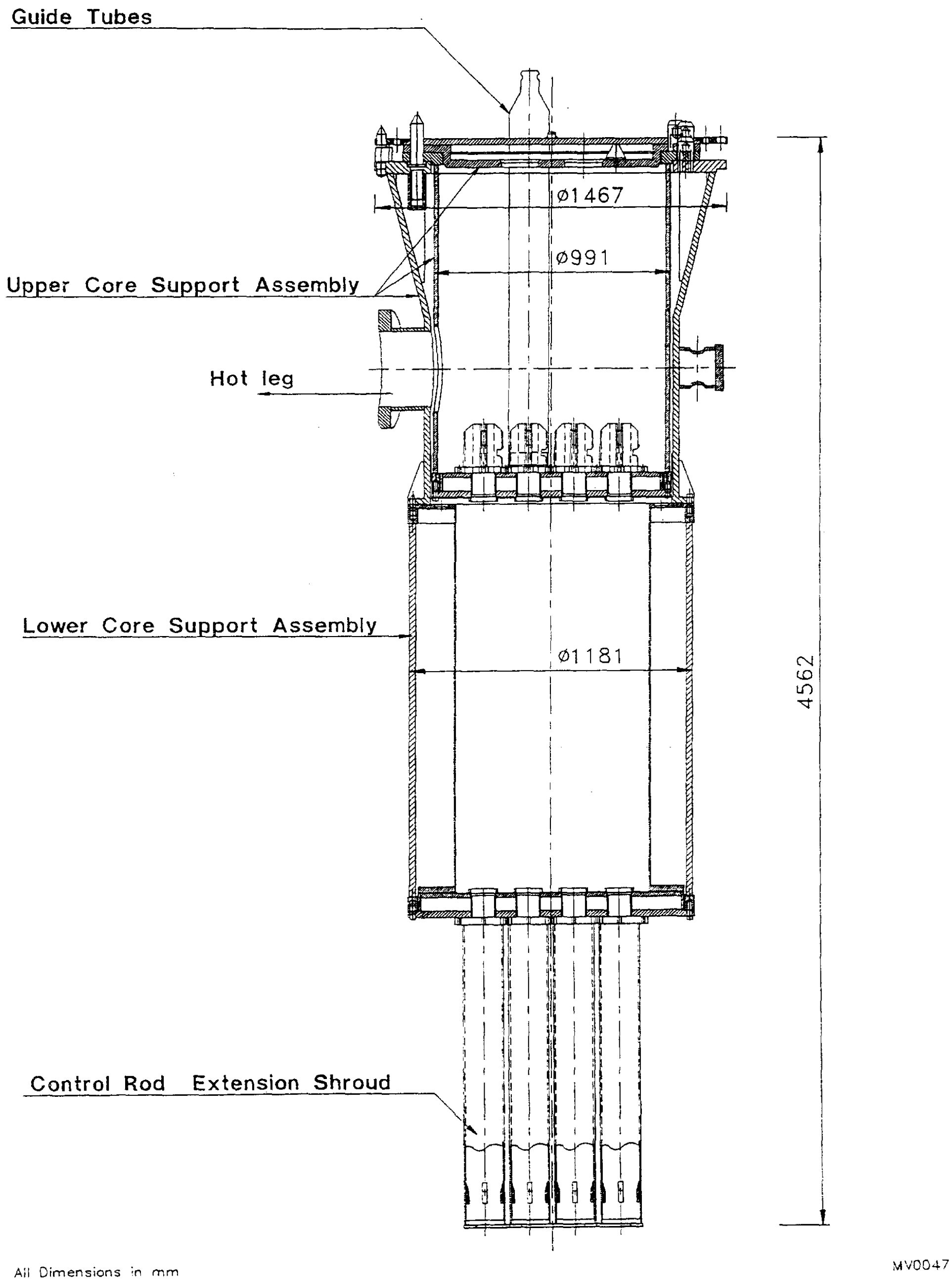


Figure 5 - General view of the BR3 reactor internals

Note : The neutron shield tank is not part of the internals

FIGURE 6

The BR3 Westinghouse Internals



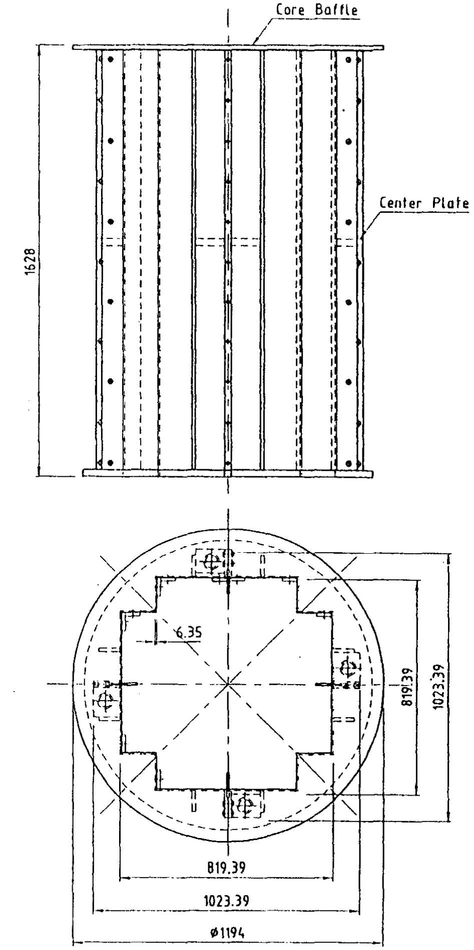
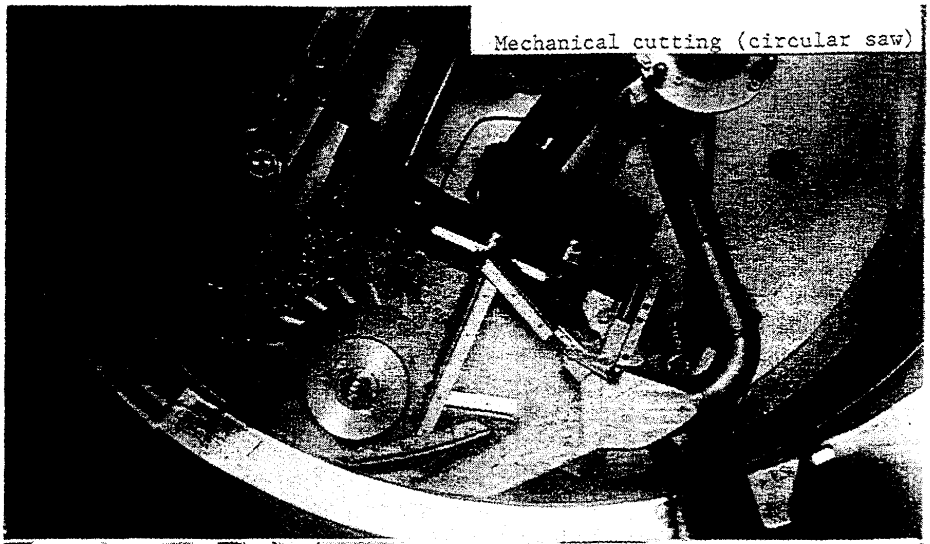
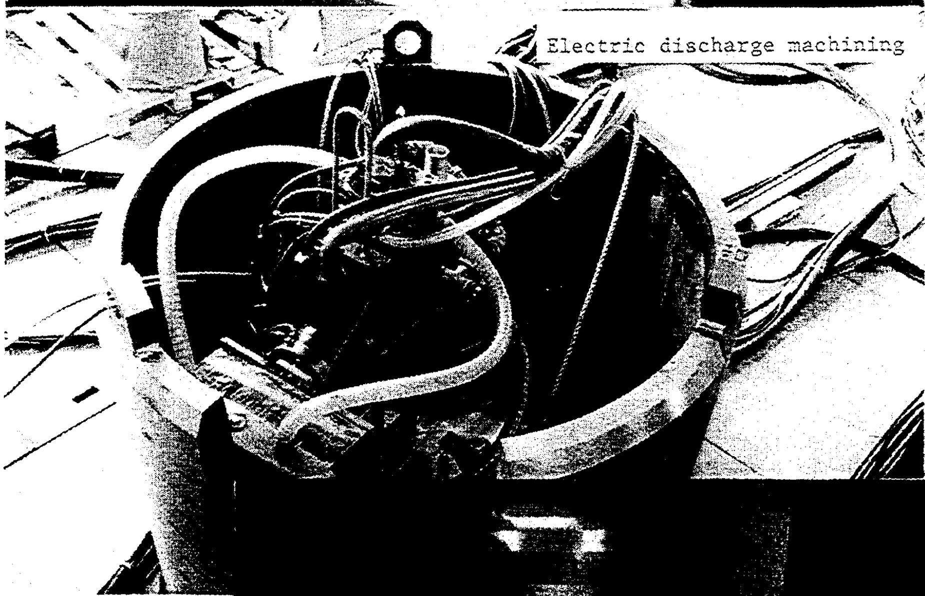


Figure 7 - View of the Westinghouse core baffle, showing the upper and lower flange



Mechanical cutting (circular saw)



Electric discharge machining

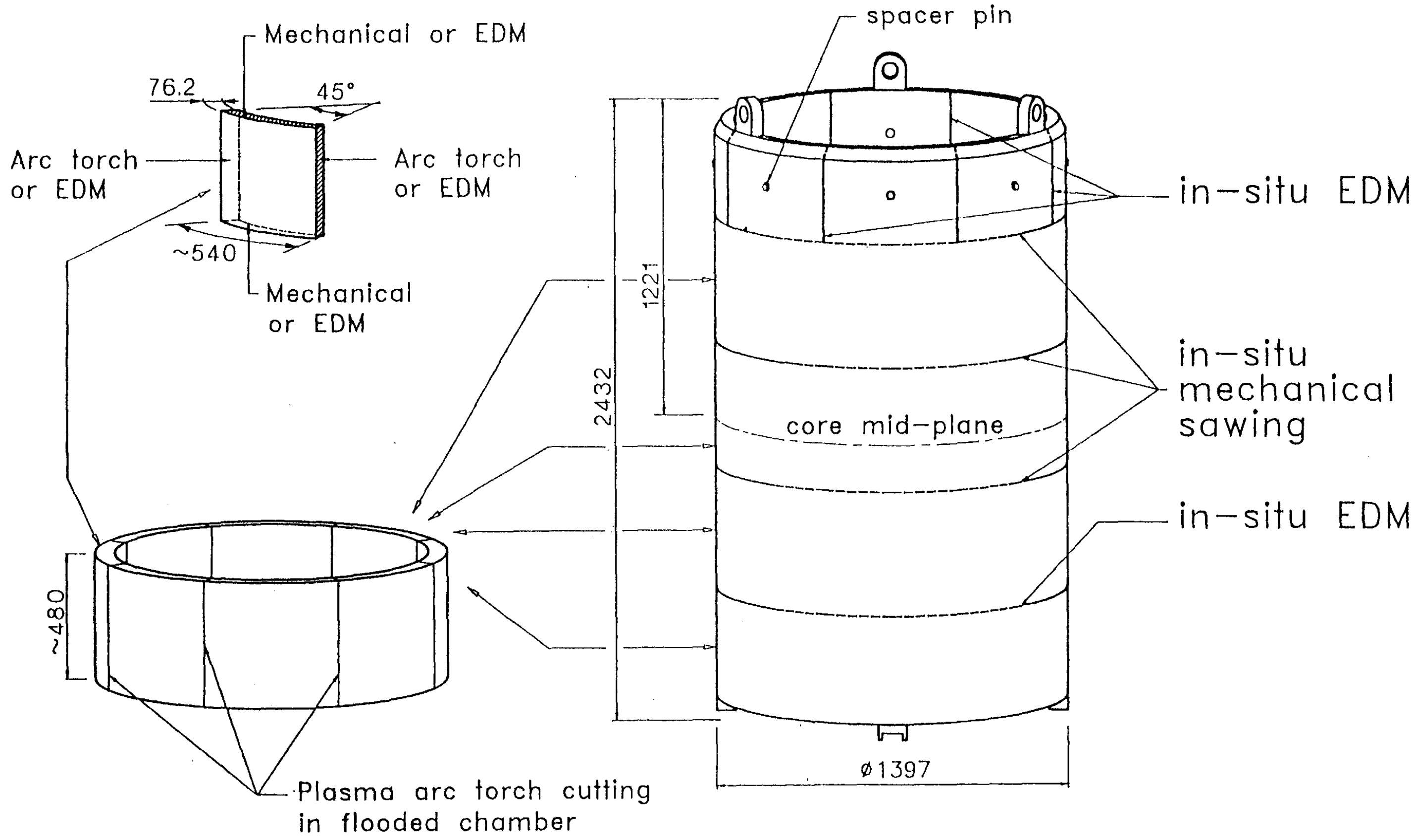
Plasma arc torch cutting

Figure 8

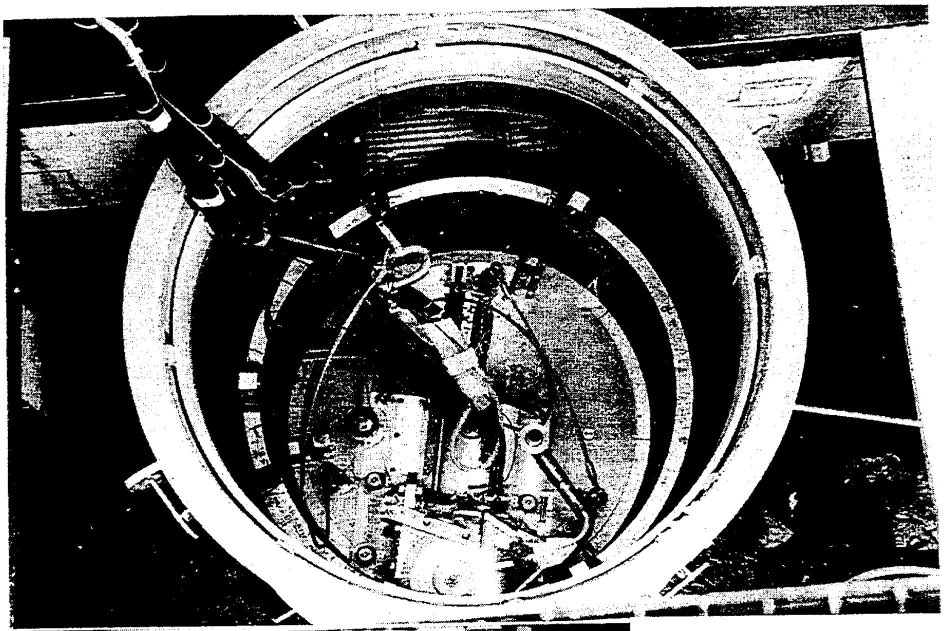
Pictures of the 3 techniques used for dismantling the thermal shield

FIGURE 9

BR3 Thermal Shield Segmentation



(All dimensions in millimeters)



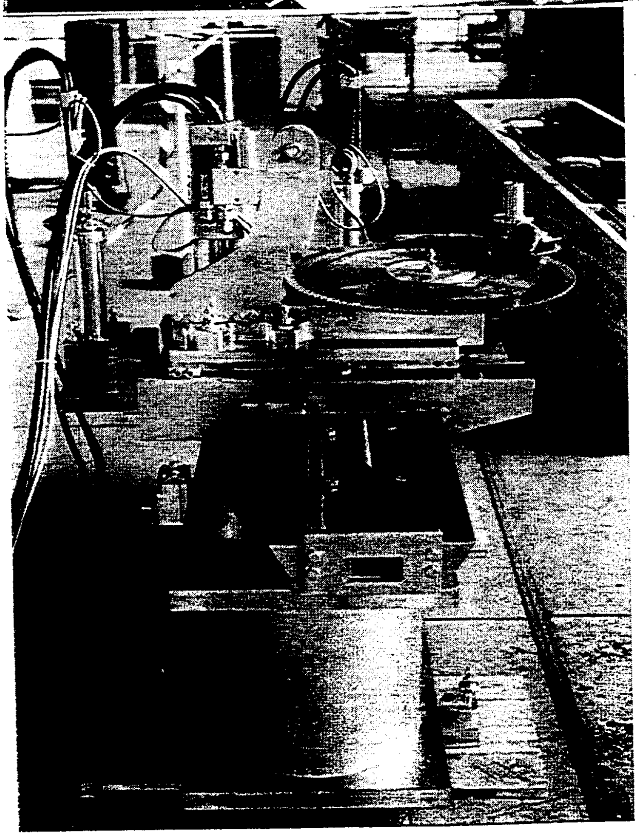


Figure 10

Cold tests on full scale mock-ups : one can see here a mock-up of the thermal shield inserted into a vessel having the same internal dimensions as the reactor pressure vessel.

Figure 11

View of the circular sawing machine used during phase 2 and 2bis, on its X-Y positioning table.

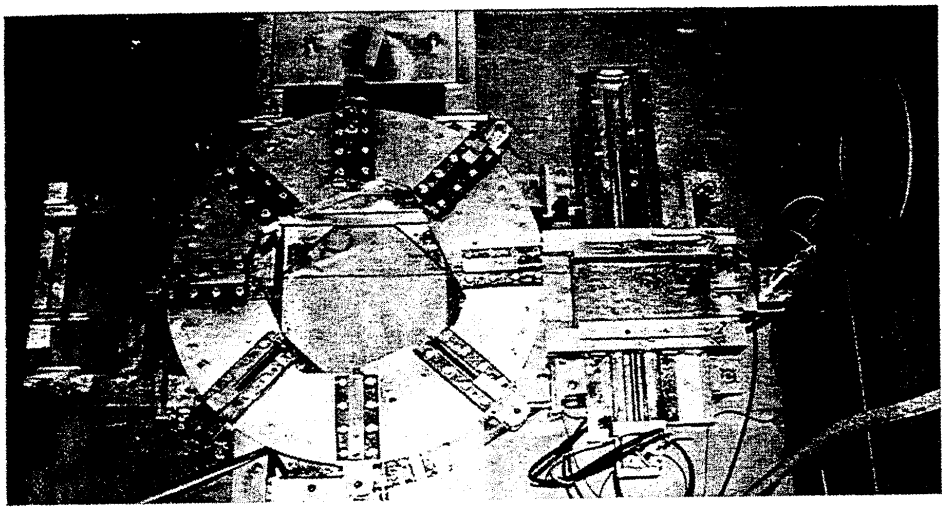


Figure 12

View of the turntable (for supporting the workpieces) and its extensions.

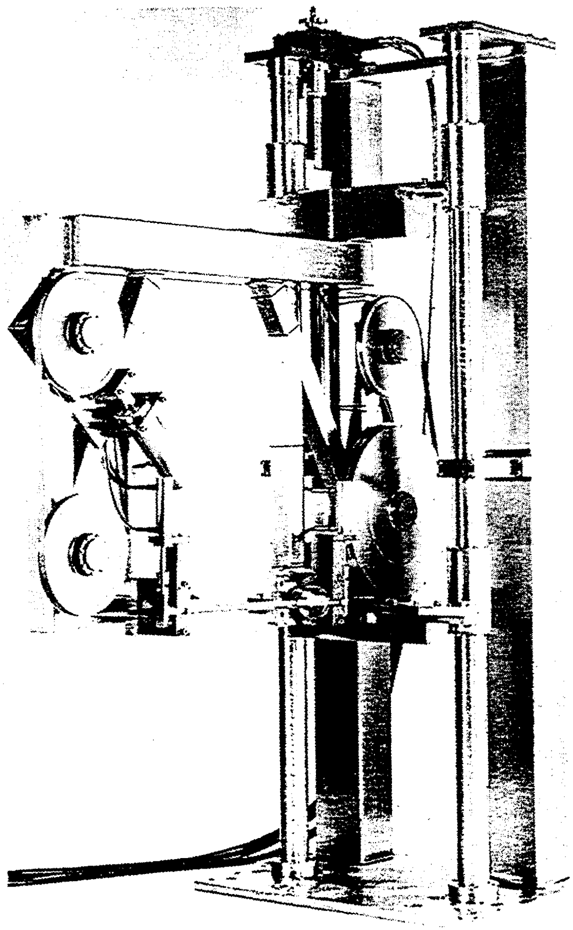


Figure 13

General view of the band saw machine

Figure 14

View of the hydraulic shears cutting pipes.

Figure 15

View of the Vulcain instrumentation basket showing the different tubes present in the upper part.

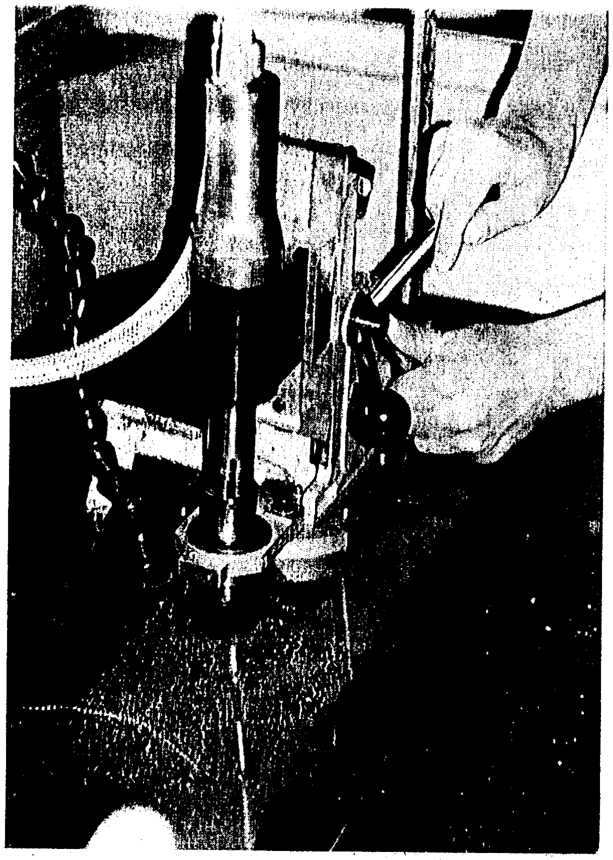


Figure 16 - View of shop tests using the core drilling.

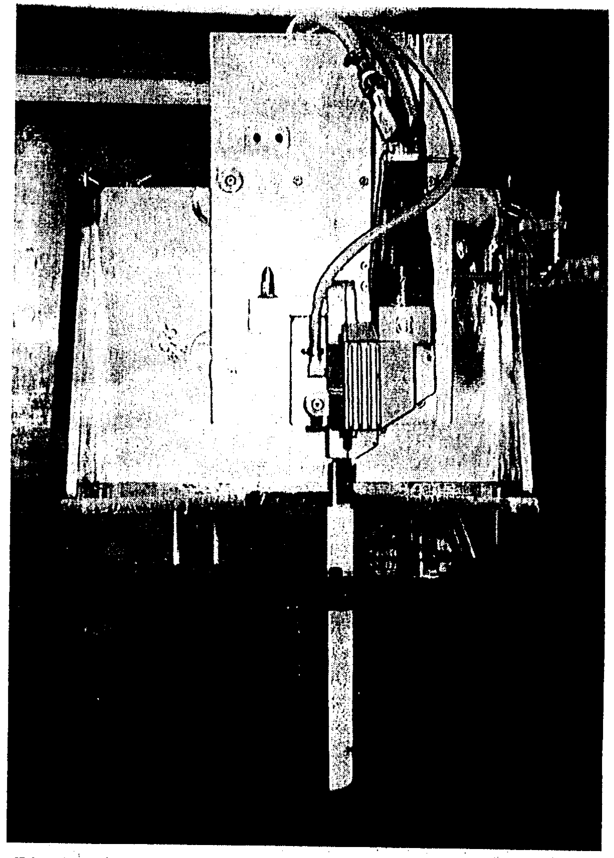


Figure 17 - View of the reciprocating saw, installed for cutting penetrating pipes of the collar, during cold trials.

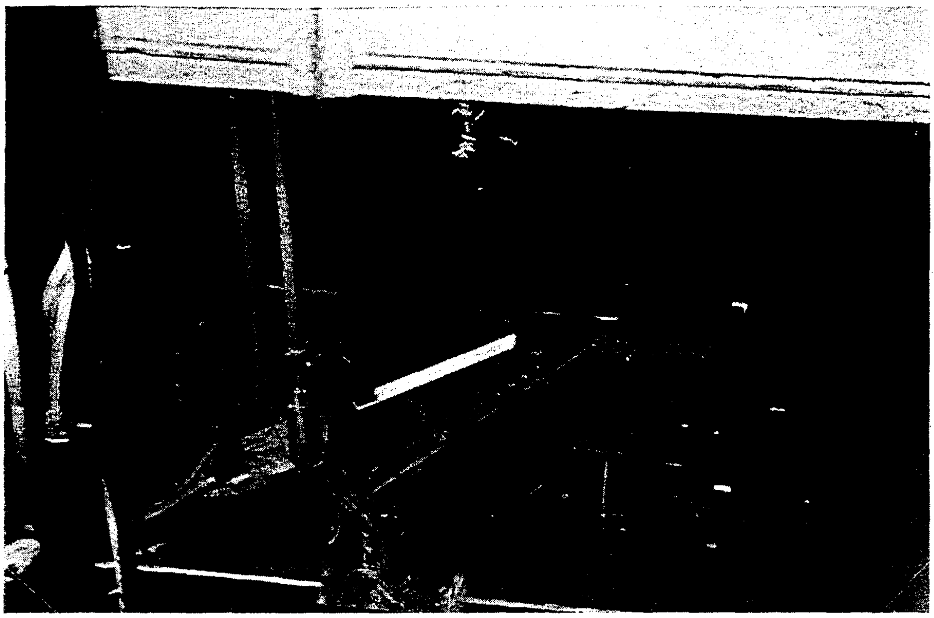


Figure 18

Underwater system for severing highly active control rods, guide tubes, etc.

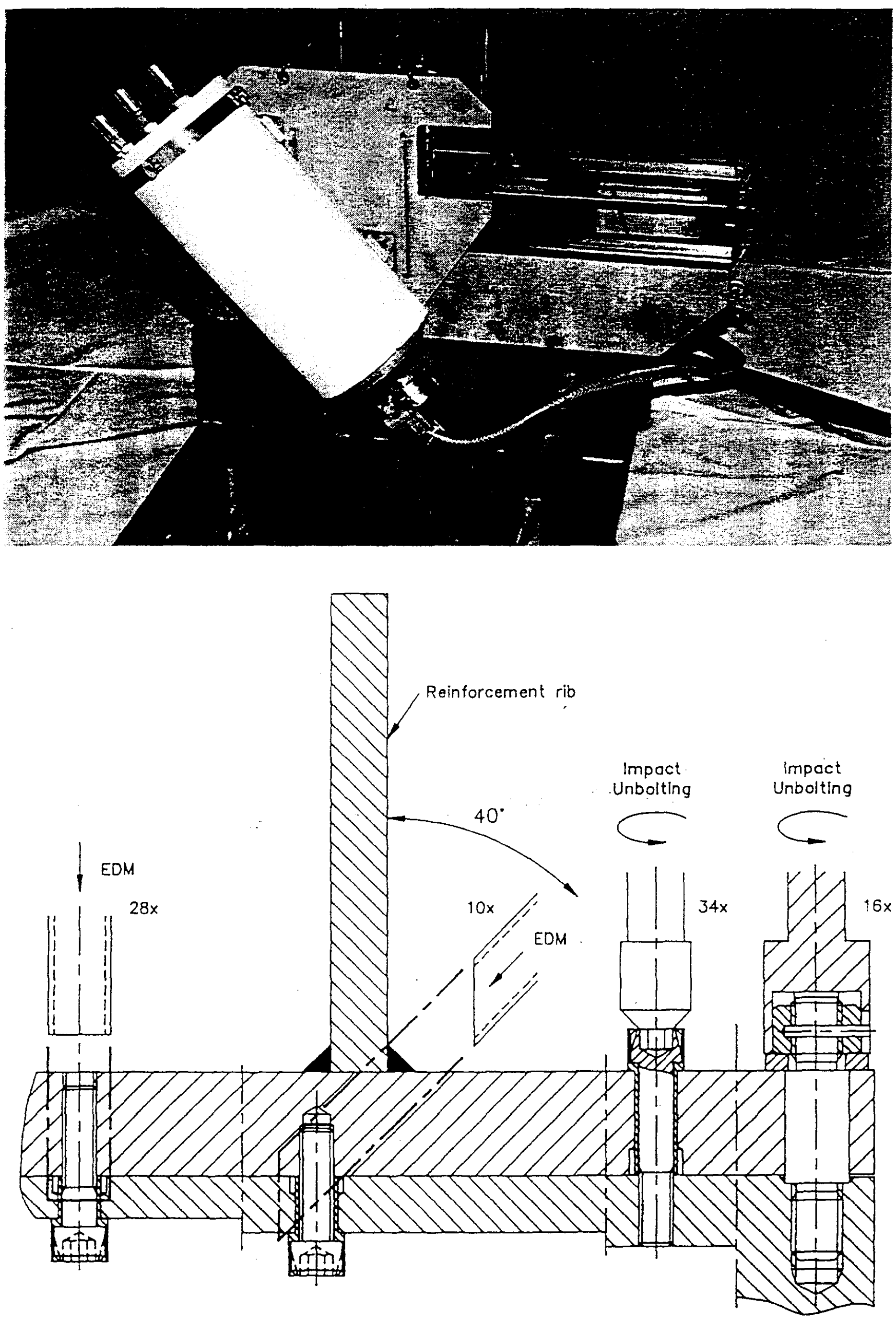
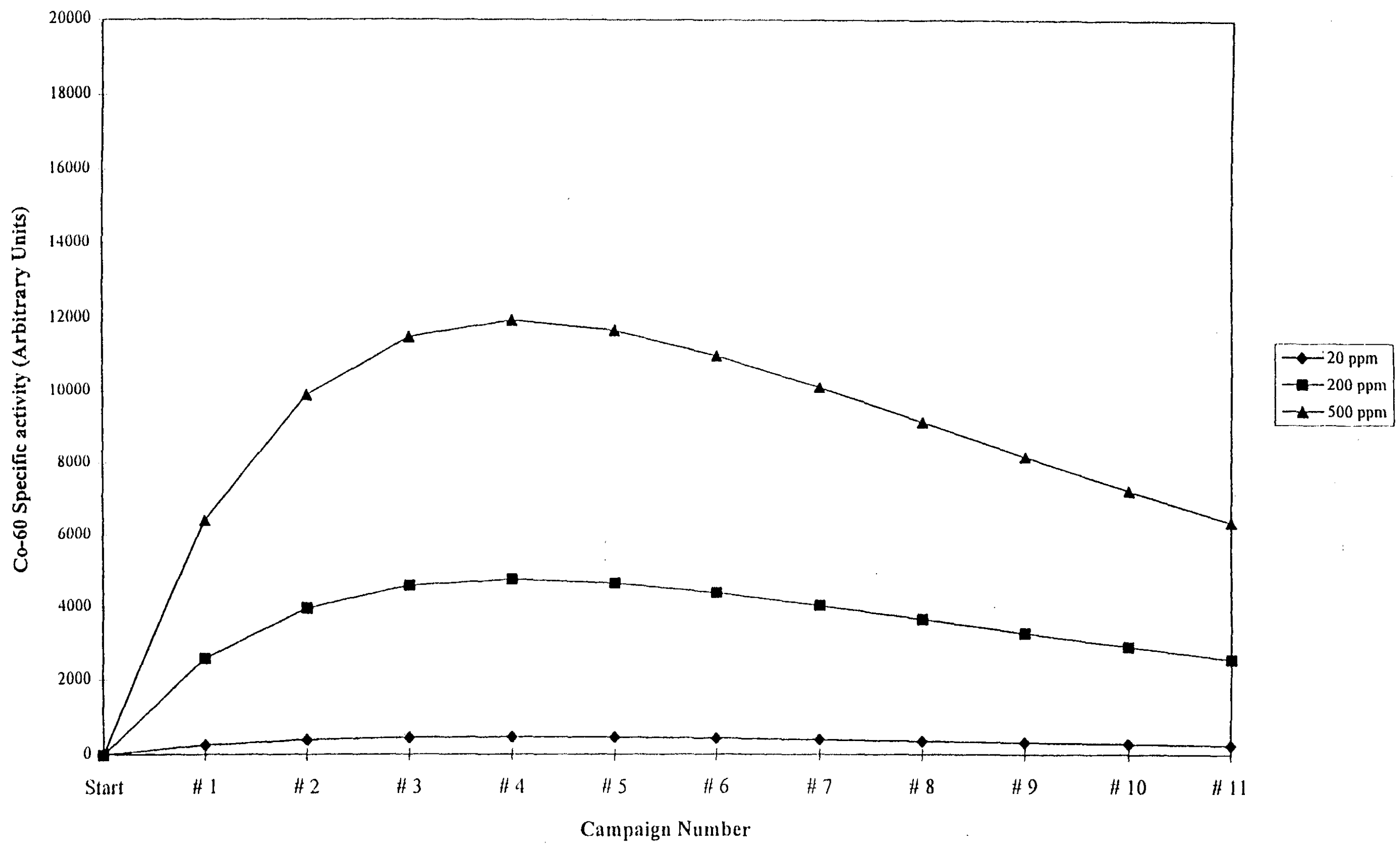


Figure 19 - EDM bolt cutting system. The scheme shows its use for accessible bolts as well as oblical perforative cut for hidden bolts.

FIGURE 20 **Evolution of Co-60 activity with the operating campaigns**



DISASSEMBLY OF THE RSSP

Collective dose for each task

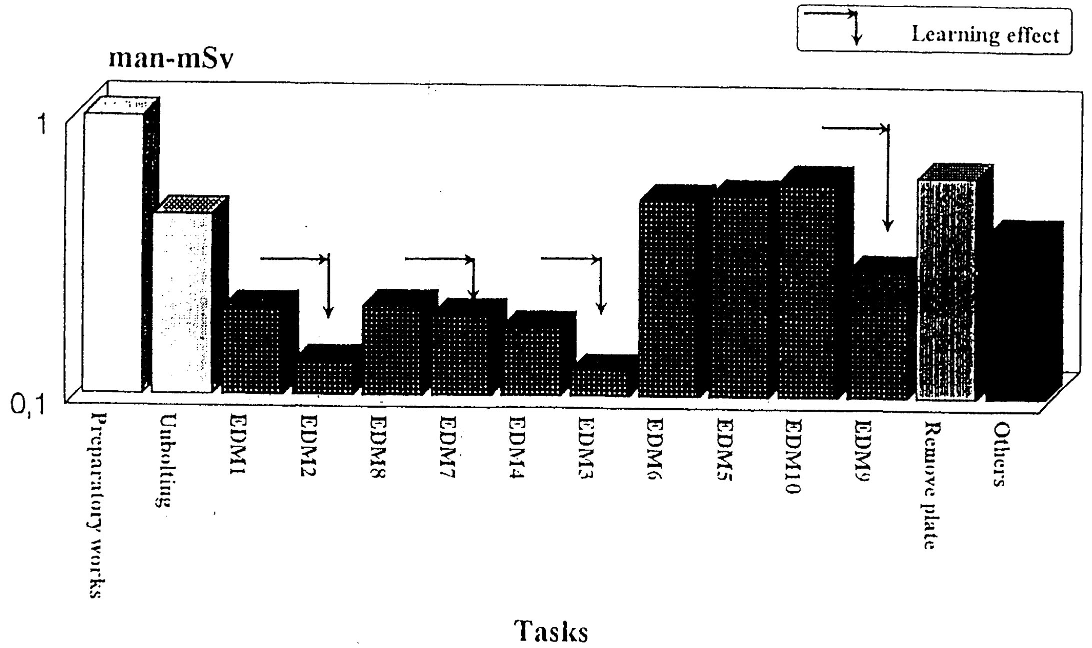
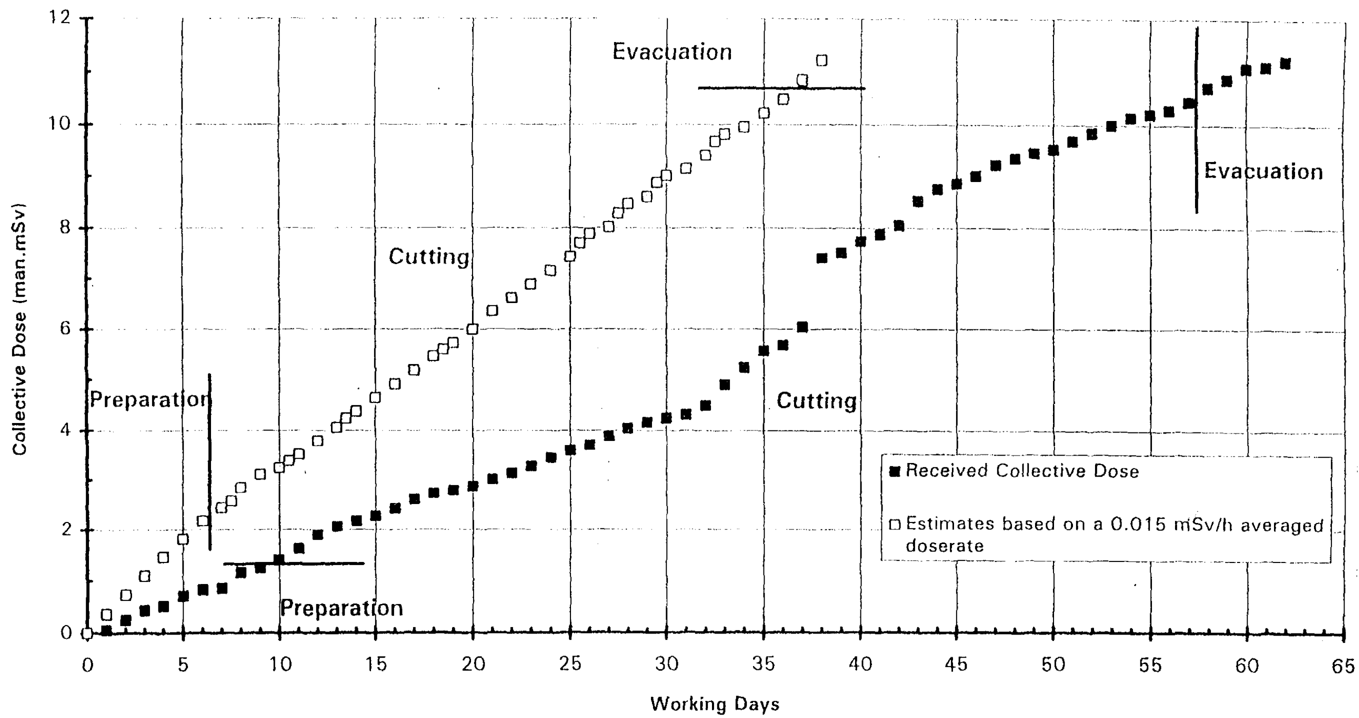


FIGURE 21

Pres2-7

Figure 22 : Comparison between estimated and received collective dose



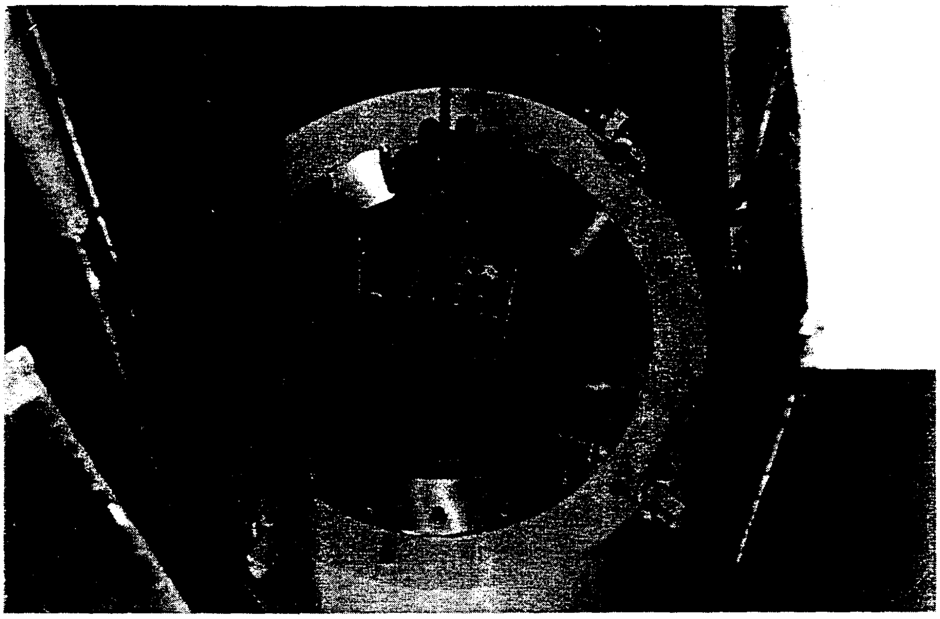
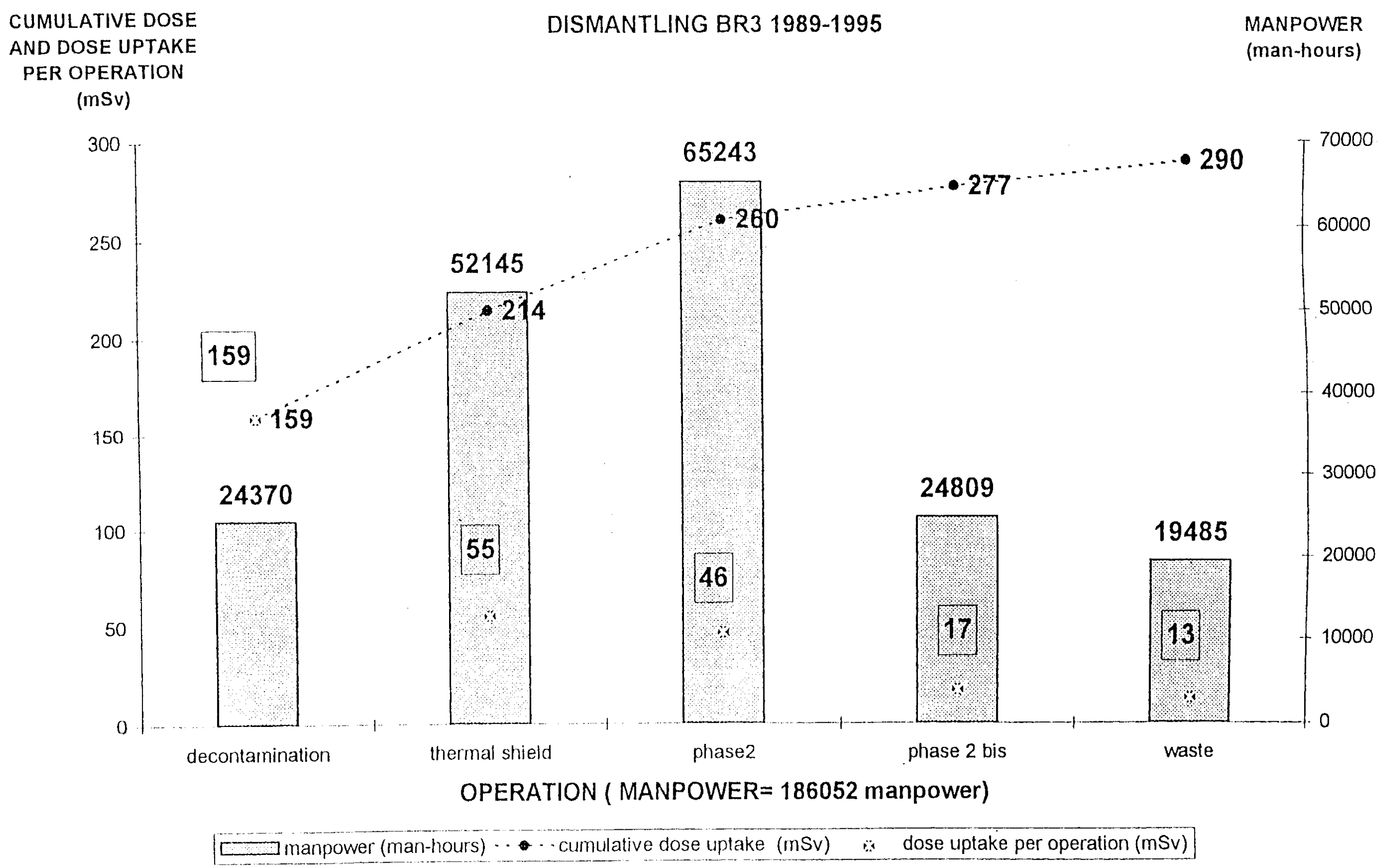


Figure 23

View of the shielded transport container for HLW during underwater loading. One can see, inside the cask, the upper flange of a 400 l drum.

Figure 24 : Summary of the main results of the project's activities



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European Commission

EUR 18229 — The BR3 pressurised water reactor pilot dismantling project

V. Massaut, A. Lefèbvre, H. Wille, H. Operschall, P. Saumon, M. Dubourg, P. Roberts

Luxembourg: Office for Official Publications of the European Communities

1998 — VII, 69 pp. — 21 x 29.7 cm

Nuclear science and technology series

ISBN 92-828-3615-0

Price (excluding VAT) in Luxembourg: ECU 13.50

BR3 is a small pressurised water reactor of 10.5 MWe output, which was started in 1962 and finally shut down in 1987. Selected in 1989 by the European Commission as one of the four pilot decommissioning projects, on-site decommissioning operations started in 1991, with the full system decontamination of the primary loop. Afterwards, the reactor thermal shield was dismantled, allowing three main underwater cutting techniques to be compared: the plasma arc torch, electric discharge machining and mechanical cutting. The next step was the remote dismantling of the remaining reactor internals, using mostly mechanical cutting methods.

Finally, internals unloaded from the reactor in 1964 and stored since then in the reactor refuelling pool were dismantled, enabling comparison between this work and that carried out shortly after shutdown. The waste management and evacuation aspects are also described.

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Research contract No. FI2D-0003-B(TT)av.1
Progress Report No.8
Period June-December 1993
January 31, 1994

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SCK•CEN, Boeretang 200, 2400 Mol
Ref. 59/94-08

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Table of contents

1. ABSTRACT

During the second semester of 1993, the following actions have been carried out :

- Dismantling of the control rods drive-shafts and associated "dash pots", which gave important information concerning the activation level at this distance of the reactor core.
- Removal of the rod shroud support plate by unbolting or destruction of solidarizing bolts. This first remote operation of phase 2 was carried out using Electro Discharge Machining and remote impact unbolting.
- Delivery, assembly and testing of the so-called turn-table for supporting and clamping the dismantling tools and the workpieces and for presenting the latter to the former.
- Complete underwater cold testing of the circular sawing machine for the horizontal cutting of the Lower Core Support Assembly.
- Underwater cold testing of the Metal Desintegration Machining, to be used as back-up technique for the circular sawing.
- Delivery, repairs and partial cold testing of the remote controlled telemanipulator for positioning the cutting tools.
- Design, fabrication and assembly of the band saw machine for cutting vertically the Lower Core Support Assembly, the reactor vessel collar (carbon steel clad with stainless steel) and for segmenting the plates and grids.
- Installation of an additional gantry crane above the reactor pool for handling the dismantled pieces and the dismantling tools.

Moreover, in the framework of the pool space occupation optimization, the internals support structure situated in the reactor pool were removed. These pieces remaining in the pool for many years were highly contaminated. They were removed, cleaned and cut into pieces in order to evacuate them from the reactor building. One of these supports (the reactor vessel collar support and shielding) was assembled on site (i.e. at the bottom of the pool) in 1964, and was never removed since then.

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A delegate member of **LAINSA** (Spain) participated also in the work packages and in the report.

2. WORK PROGRAMME

The programmed works for the period were the following ones :

- Removal of the internals supporting structures from the reactor pool. This operation concerned the LCSA support structure as well as the vessel collar and instrumentation basket support and shielding. The shielding is constitued of a cylindrical assembly, made of 3 cylinder subassemblies, and containing about 6 T of lead.
  This part of the programme has been completed (see chapter 3.8.).

- Cold testing on partial full scale mock-up of the horizontal cutting system (circular saw and MDM). The circular saw cold test has been carried out completely, the results are given in chapter 3.4.1.
  The MDM cutting, newly developed for this purpose, showed still some problems, to be resolved before application on radioactive pieces. This part of the programme is described in chapter 3.4.2.

- Cold testing of the telemanipulator and training of operators. This part of the programme has been carried out only partially due to repetitive problems encountered with the hardware and software of the telemanipulator and its control system (see chapter 3.3.).

- Assembly and cold testing of the so-called "turn-table" : this part of the work was carried out at the same time as the cold testing of the concerned dismantling tools. Step 1 (for the upper internals dismantling) and step 2 (for the horizontal cutting of the LCSA) were intensively used and tested. Step 3 (for the vertical cutting of the LCSA and collar) has been delivered and tested on a simplified model (see chapter 3.6.1.).

- Final cold testing of EDM for removing bolts and operation "in-situ". The complete removal of the rod shroud support plate has been carried out without major problems. This operation is described in chapter 3.2.1.

- Desolidarisation of the reactor collar from its instrumentation basket : this part of the work has been partially achieved (see point hereabove), the completion of the work being foreseen early in 1994 (see also chapter 3.2.2.).

- Design, fabrication and assembly of a band saw machine for cutting the LCSA and the collar vertically and for segmenting the plates and grids.
  This work has been completed (see chapter 3.5) and the machine is ready for cold testing on scale 1:1 mock-up early 1994.

Some other operations, not directly programmed for this period, were carried out. For instance, the complete dismantling and evacuation of the control rods drive shafts and associated "dash pot" were executed in the air (see chapter 3.1.).

3. PROGRESS AND RESULTS

3.1. Removal and dismantling of the control rods drive shafts and associated "dash-pots"

In order to liberate the access to the rod shroud support plate (see chapter 3.1.2.), the control rods drive shafts situated in their guide tubes, had to be removed and evacuated. These equipments were not too highly activated, the control rods being withdrawn from the reactor during operation. Nevertheless, due to their quite complicated geometry, the drive shafts and the associated dash-pots were not completely decontaminated and cleaned during the full system decontamination carried out in the framework of phase 1 of the project. Some radioactive hot spots were measured under water, after removal of the different drive shafts (max. radiation level = 10 mSv/h, contact).

However, after a first cleaning by high pressure water jet, the radiation level was low enough to allow hands-on dismantling with quite simple shielding material (e.g. lead mats). The drive shafts, also called "spaghetti bundles" after their geometrical form (i.e. 7 long flexible bars), were cut by grinding into pieces with a length shorter than 60 cm. They were then stored into a standard 200 l waste drum. The dash-pots (see picture 1) presenting the highest radiation level (4 mSv/h) were sent to the existing decontamination test loop where they underwent decontamination using the CORD process and ultrasonic cleaning. This operation removed the principal hot spots and brought the dose rate level below the 2 mSv/h contact dose rate limit (200 mR/h). This is a very important result : indeed, above this limit the waste is considered as medium or high level waste requiring special treatment by the belgian national organism for radioactive waste disposal ONDRAF/NIRAS. Below this 2 mSv/h limit, the waste is considered as "standard" waste, i.e. ondergoing standard conditioning operations and then evacuated at lower cost.

The result concerning the drive shafts was important because it meant that the activation at that level of the reactor (i.e. 700 - 1500 mm above the top of the fuel assemblies) is sufficiently low and that the effect of the primary loop decontamination operation is sufficiently high to allow all metallic material situated at and above this level to be evacuated as standard low level waste.

The conclusion of this operation is then double : first, it showed a quite unexpected result of the primary loop decontamination, allowing re-categorization of waste arising from the internals dismantling. Second, it implies that the estimated amount of high level waste coming from the dismantling of the reactor internals can be significantly reduced. This will also require to sort the cut pieces on radiation level after their dismantling, for the internals situated at more than 500 mm from the core top-plane.

3.2. Desolidarisation of the collar and instrumentation basket (Task B.4.3.)

3.2.1. Removal of the rod shrouds support plate using EDM

The two techniques required to remove the screws on the rod shroud support plate (the impact unbolting and the Electro Discharge Machining) are well known and under control. The main part of our last cold tests was dedicated to define precisely the parameters of the positioning when using the X-Y table in order to bring the EDM head at the right position to remove a screw.

Installation

During a few days, the work consisted in bringing the materials through the airlock and in installing them in position. When the platform foreseen to go on the guide studs of the reactor pressure vessel was assembled on the operating deck, it was then installed on the guide studs at its working position, about 2 meters above the reactor pressure vessel upper flange. The installation of the ladder giving access to the working level, required a lot of work because support pieces had to be welded on the reactor pool wall lining. The rest of the installation works consisted mostly in reproducing the configuration of the installation that we had during the last phase of the cold tests, i.e. :

- on the operating deck level :
  - the EDM control panel
  - the control unit for the pumping device (particle suction through the electrode)
  - the hydraulic pump for the clamping of the X-Y table
  - the underwater video control unit;

- on the platform level :
  . the pumping device
  . the filtration unit
  . the hydraulic unit which drives the EDM head.

Later, it was decided to add a video monitor on the platform in order to help the operator for some precise works.

Execution of the work

As stated in the last progress report [1], some preparation works were needed before using the X-Y table on the rod shrouds support plate. The removal of the two "trumpets" by unbolting the eight M12 fastening bolts was carried out without problem. Then, the bending of the connecting pieces of the thimble tubes (which were in the "trumpets"; see [1]) could be performed. As it was calculated, the force needed on the tool (a strong tube) at the working platform level to bend the connecting pieces could be applied by hand.

Then, a first unbolting sequence was started. It concerned the M12 bolts which fastened the Corthals Joints under the rods shroud support plate. Only one of the twenty-six bolts broke during unbolting but this will have no influence on the removal of the plate, the movement being parallel to the axis of the bolt. It was also possible to collect the unscrewed bolts by using a long gripping tool.

Afterwards, the work with the X-Y table and the EDM process to cut the screws which were impossible to unbolt due to their "top-down" position, took about twenty working days. This job was divided into ten operations. Each operation consisted in a long sequence of elementary tasks. These sequences were defined in order to minimize the intervention of the operator from the lower platform on the EDM head and on the X-Y table (ALARA). In fact, each operation represented one position of the X-Y table on the rod shrouds support plate.

During this cutting phase, ten bolts were cut oblically and twenty-eight vertically. During the cutting period, no operator was present on the platform, only one operator stayed on the operating deck in order to conduct the EDM process and to check regularly the filling of the filters. When the pressure downstream the filters rose to 2 bars, they were changed by one operator on the platform itself.

This work did not imply an important dose to the operators because it was a very fast operation (only a few minutes) and the collected material was not very active (the maximum dose rate at the surface of the filters presenting the highest radiation level was 1 mSv/h).

When all the screws were cut by EDM, a thin instrumentation tube, traversing the rod shrouds support plate, had to be cut. This work was also performed using the EDM head, although it was not foreseen during the cold test. The last desolidarisation work was the impact unbolting of the M18 secured bolts between the plate and the reactor vessel collar. This unbolting is more difficult than the other one due to the bigger resistance of the bolts and to the type of securisation (traversing pin, Ø 4 mm). Indeed, in this case, a long sequence of bolting-unbolting-bolting ... was needed before the bolts broke at the securing pin level.

Afterwards, all equipment, except the platform, was evacuated and the rod shrouds support plate was removed using three chains and three hooks (one in each handling lug of the plate). Due to the fact that it is very difficult to lift the plate precisely vertically, some assembly dowels gave a little resistance against the desolidarisation. However, after about one hour of work and inspection with the underwater video camera, the rod shrouds support plate could be removed. The first visual inspection (afterwards) showed that all the screws were completely cut by the EDM operations. This meant that all the positioning phases were well performed and that the wear of the electrodes was not deviating from the cold test values. The plate was then flushed and placed in the plasma arc torch flooded chamber for temporary storage, waiting for the band saw machine to be cut into pieces.

Results

During this work, 88 bolts or screws were removed or cut remotely and under water (10 were cut oblically and 28 vertically using EDM). The parameters of the EDM screw cutting are given in the two tables at the end of this chapter.

Oblical cutting and vertical cutting are separated because they are very different. Indeed, the vertical ones gave no problem and the positioning was relatively easy. On the other hand, the oblical cutting is much more delicate : during the positioning, the distance between the position of the axis of the screw and the point where we had to start cutting was fundamental.

If it is too long, the screw will not be completely cut and if it is too short, the screw will be cut at a too high level and the desolidarisation will not be possible. Also, the oblical cutting is more difficult because it releases a drill core which can perturb the cut at its end. The suction is more difficult when the oblical cutting is used. Indeed, the oblical cutting requires electrodes which are not axisymmetric, the cut starting in corners of the plate (corner between the thick plate and the reinforcement ribs of the plate).

Table 1 presents the vertical cutting results. Some parameters, as the electrode wear, could be measured because the electrodes were not strongly contaminated. But when the same electrode was used for cutting more than one bolt, the total wear is simply divided by the number of the cut bolts. So, each bolt cut by the same electrode, by hypothesis, has the same wear in the table.

Two cuts gave little problems. During cut 8.1., a cable inside the EDM control cabinet burnt. The cut was interrupted in order to replace it. Cut 6.4. was very disturbed due to interferences between the electrode support or the suction line and the workpiece.

The mean value is shown at the bottom of the table. The electrode wear (about 17 to 19 %) is quite high. The electrode material used (copper) is probably not the best one to cut stainless steel but it has the main advantage that it is easily machined (the electrodes were fabricated on-site in the workshop of the plant).

Nevertheless, the mean speed value (given in mm^3/min of eroded base material) is quite close to the value (210 mm^3/min) given in the litterature for classical EDM on stainless steel carried out in industrial workshops and in a dedicated dielectric fluid [3].

Table 2 presents the same type of results for the oblical cutting. For each bolt a new electrode was used. The performances here are not as good as for the vertical cutting, due to the different reasons mentioned above. Moreover, the speed was intentionally reduced in order to avoid any blocking of the electrode at the end of the cut. Indeed, the back-up solution in case of blocking (removal of the head, recuperation of the electrode, positioning of the head and starting of a new cut in the same hole) would take much more time than the additional time required for a slower cut but without major problem.

Figure 1 presents the results regarding the filters. The filter cartridges were exchanged when the pressure drop amounted to 2 bar. Ten filter sets were used. Each set is composed of two cartridges (10 inches long). The filter cartridges used are of the depth filter type, with a mesh size of 1 μm. The filter medium is polypropylene which can be burnt for waste conditioning and volume reduction.

The volume collected in each set is based on a calculation made on the number of holes carried out with each set and the depth of each hole. The volume is thus a theoretical one and it does not take into account that the suction is not perfect and that all particles were not trapped.

The mean value amounts to 55 cm^3/set. The tenth set is not taken into account for calculation of the mean value because it was not full at the end of the work. The total volume of the cut material amounts to 524.5 cm^3 for a total filter volume of 15836 cm^3. This means a ratio waste volume/cut material of 0.033. The amount of material collected is about 4.2 kg or 210 g/filter.

Summarizing the results we can thus say that :

- cutting a bolt vertically by EDM took in average 0.39 working day (total operational time) and resulted into a dose of 0.058 man-mSv
- cutting a bolt oblically by EDM took in average 0.8 working day and resulted into a dose of 0.121 man-mSv.

Table 1 - Vertical cutting results

Operation nr.	Bolt nr.	Cutting time (minutes)	Problems occurred	Length wear mm	Length wear %	Speed mm/min	Speed mm³/min
1	3	71	no	8.7	17.40	0.70	168.14
	2	72	no	8.7	16.57	0.73	174.10
	1	62	no	8.7	16.11	0.87	207.95
2	3	66	no	9.3	17.22	0.82	195.35
	2	68	no	9.3	17.38	0.79	187.85
	1	63	no	9.3	18.24	0.81	193.28
8	3	60	no	8.5	16.04	0.88	210.90
	2	50	no	8.5	16.04	1.06	253.09
8	1	58	yes	8.5	16.50	0.89	212.00
	x	70	no	8.5	15.18	0.80	191.01
7	3	75	no	8.5	16.67	0.68	162.36
	2	90	no	8.5	17.00	0.56	132.64
7	1	74	no	8.5	16.04	0.72	171.00
	x	71	no	8.5	16.35	0.73	174.87
3	1	66	no	8.5	15.18	0.85	202.58
	3	62	no	8.5	15.18	0.90	215.65
4	3	65	no	9.5	16.67	0.88	209.37
	1	77	no	9.5	18.10	0.68	162.79
6	1	70	no	9	16.98	0.76	180.78
	2	62	no	9	16.67	0.87	207.95
	4	xxxxxx(*)	xxxxxx(*)	xxxxxx(*)	xxxxxx(*)	xxxxxx(*)	xxxxxx(*)
5	1	61	no	9	17.31	0.85	203.53
	2	64	no	9	17.14	0.82	195.86
	4	66	no	9	16.67	0.82	195.35
10	1	74	no	8.5	17.00	0.68	161.32
	3	65	no	8.5	16.50	0.79	189.17
9	1	86	no	10.5	20.00	0.61	145.75
	3	78	no	10.5	18.42	0.73	174.48
MEAN		**68.37**		**8.93**	**16.83**	**0.79**	**188.12**

(*) <u>NOTE</u>

Cut repeatedly interrupted (see the text).

Table 2 - Oblical cutting results

Operation nr.	Bolt nr.	Cutting time (minutes)	Problems occurred	Length wear mm	Length wear %	Speed mm/min	Speed mm^3/min
4	2	132	yes	16	19.51	0.62	148.32
3	2	117	no	14	17.07	0.70	167.34
5	3 5	187 96	yes no	12 15	14.72 18.07	0.44 0.86	104.06 206.43
6	3 5	159 109	yes yes	13 18	16.67 21.69	0.49 0.76	117.13 181.81
9	2 4	149 123	no no	16 21	20.00 25.61	0.54 0.67	128.19 159.17
10	2 4	176 176	no no	13 18	16.25 22.50	0.45 0.45	108.53 108.53
MEAN		**142.4**		**15.60**	**19.21**	**0.60**	**142.95**

3.2.2. Desolidarisation

After the removal of all the pipes and equipments in the upper part of the instrumentation basket using the hydraulic shears, the collar has to be "cleaned". Indeed, the collar presents penetration tubes which are too thick or not accessible for the hydraulic shears. Therefore, another cutting equipment was chosen to cut these penetrations as close as possible to the inside wall of the collar. This equipment consists of a pneumatic reciprocating saw adapted to work under water and supported on a, for this purpose, designed frame [1].

The equipment with all its parts (clamping device, feed screw mechanism, ...) was fabricated and tested on a 1/1 scale mock-up of the collar (see picture 2). Indeed, some penetrations were also foreseen on the mock-up and the positioning from the lower working platform, placed on the reactor vessel guide studs, didn't give any problem.

This operation in the reactor building is foreseen for the beginning of '94 just after the instrumentation basket dismantling and it will take about 3 days work for 3 persons.

3.3. Dismantling of the instrumentation basket and the Upper Core Support Assembly (UCSA) by means of a telemanipulator and adapted cutting tools (Tasks B.4.2 - B.4.3.)

The instrumentation basket and the UCSA are mainly composed of tubes and pipes, and their specific geometries and shapes were described in detail in the previous reports [1][2].

It was foreseen to dismantle them by cutting the pipes by means of an hydraulic jaw cutter and a reciprocating saw for the largest diameters. Cold test of both tools were carried out and previously reported.

For the positioning of the tools and for the recuperation of the cut pieces, it was foreseen to use an underwater remote controlled, hydraulically driven telemanipulator.

The telemanipulator was delivered in July 93, but due to a lot of hardware and software malfunctions, the effective use of the telemanipulator for training and testing purpose was reduced to a few days. Moreover, different hardware problems have put the reliability of the manipulator in doubt.
Therefore, studies and tests have been carried out to be able to execute the operations "by hands" using adapted long handling tools.

Nevertheless, some brief cold trials have been carried out with success on a scale 1:1 mock-up using the telemanipulator (see picture 3). Small pipes, up to 44 mm O.D. were cut with the hydraulic jaw cutter, and the spray-in collector of the UCSA (see figure 2) was cut using a reciprocating saw and a specially developed clamping system.

For the actual dismantling of the instrumentation basket and the UCSA, both strategies (with and without telemanipulator) will be followed, if possible.
It will allow to compare the two strategies, from a radiological and financial (time spent) point of view. It will also allow to be almost independent of the telemanipulator poor reliability, having always a back-up solution.

Moreover and following the general strategy set up for the internals dismantling [2], the highest levels of the instrumentation basket (situated just under the rod shroud support plate) will be dismantled using the hydraulic jaw cutter fitted to a long handling tool. This will allow to carry out the remaining operations.

3.4. Dismantling of the Lower Core Support Assembly (LCSA) (Task B.4.1. - B.5.1.)

3.4.1. Horizontal mechanical cutting with circular saw blade (Task B.4.1. modified)

Introduction

The strategy and process for the horizontal cutting of the lower core support assembly have been described in a previous progress report [2]. Also the manufacturer's shop tests of the saw blade cutting equipment have been dealt with previously [1]. Since then, the equipment has been delivered to CEN•SCK and further in-house tests have been performed to verify the equipments performance and reliability and to train the operators. The tests were performed on a simplified actual size mock-up, representative of the lower core support assembly. They took place in the machine hall of the plant from September until November 1993.

Cutting tests on mock-up

It is necessary to perform "cold" cutting tests before one cuts the lower core support assembly in the controlled area. The "cold" cutting tests serve to compare actual and anticipated performance of the cutting equipment and to verify its reliability in a substitute environment. They also serve to get a feel of the equipments cutting characteristics and to train the operators. Furthermore, any remaining problems can be solved during these "cold" tests. It would be much more difficult and costly to solve them during the execution of works in the "hot" area.

A mock-up, representing the lower core support assembly, was made and placed on the turn-table in a test tank-(Picture 4). On this mock-up four types of mechanical cuts were tested. They were representative of the following cut levels on the lower core support assembly (Figure 3):

.level I	-	between the lower core support plate and the core barrel,
.level II/III	-	the core barrel,
.level IIbis/IIIbis	-	the core baffle,
.level IV/V	-	the lower core support barrel.

The mechanical cutting equipment (Picture 5) consists of a base, the X and Y swallow-tail positioning guides, the circular saw blade with its drive, a handling device and turn-table fittings.

The "cold" tests of this equipment were done over a time-span of 31 working days. A total of approximately 22 man-weeks were spent on them.

Each cut to be performed was first studied using a CAD (Computer Aided Design) system before the actual cutting started. An example of a resulting cut plan is shown in Figure 4. The advantage of cut planning using CAD is that alternative ways of cutting can be analysed more effectively. This will enable to optimise the estimated execution time and dose rate when planning for the actual horizontal cutting of the lower core support assembly in the controlled area.
Four different saw blade types were tested during the cold tests (Table 3). The *Ø600mm/5mm pitch* saw blade was successfully used for cutting the core baffle, a structure of thicknesses as low as 1.6mm. The *Ø400mm/19mm pitch* saw blade is to be the standard saw blade for all other cutting. However, it can only protrude 95mm into the work piece. If deeper cuts are required, the *Ø600mm/19mm pitch* saw blade must be used. This can protrude to approximately 200mm into the work piece. For certain cut levels the maximum cut depth is further reduced due to interferences between the turn-table clamps and parts of the cutting equipment, e.g. at level I the maximum depth of cut for the Ø400 and Ø600 mm saw blades is respectively 50 and 155 mm. A *Ø600mm/9.6mm pitch* saw blade is provided to be used for deeper cuts of thinner materials, e.g. tubes. Where the lower core support assembly is only 25mm thick, the latter blade is the only one for which there is no limit on the depth of a pass on the exit side of a cut (see Figure 5). Indeed, there will always be about 2.5 teeth in the cut no matter what the depth of the pass is. This can substantially reduce the cutting time since all exit sides of the passes making up the cut can be cut with the final pass.

Table 3 lists the values of the cutting parameters which are to be used for cutting the lower core support assembly. These values are based on the results of the cold tests of the mechanical cutting equipment on the mock-up.

Based on the times taken to perform different types of operations whilst cutting the lower core support assembly mock-up, time estimates for set-up and manipulations during mechanical cutting of the lower core support assembly can be obtained. These estimates are listed in Table 4.

Figure 4 shows a cutting plan for a cut on the mock-up. This cut is representative of a level IV cut on the actual lower core support assembly. The information contained in this figure (length of cuts and returns), in Table 3 (cutting speed) and in Table 4 (operation time), can be joined to yield an estimate of the time needed for this type of cut. It amounts to approximately three six-hour[1] cutting days (Table 5) plus 1 day for set-up (Table 4). Here, no account has been taken of filter or saw blade changes or any unforeseen problems. The time information can further be used for calculating dose estimates for radiation protection optimisation.

Some problems were encountered during the cold tests. Some of them and their solution are described next.

It was essential to retrieve as much of the chips as possible. Indeed, one of the reasons for selecting the circular saw for horizontal cutting was the large size of the swarfs, which makes their collection easier. However, problems arose with the swarfs collection and suction system. By now, these have majorly been solved through modifications of the system during the cold tests, i.e. better guidance of the swarf collector, improved suction flow distribution in the swarf collector and optimisation of the position of the brushes around the saw blade.

Hydraulic problems arose due to the incompatibility of the bearing material used in the hydraulic pump of the positioning circuit with the water glycol mixture used as hydraulic fluid. The gear type hydraulic pump has been changed for a multiple piston type.

The positioning and return speed did not match the demanded one and was increased by the installation of additional valves.

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[1]One working day normally amounts to eight hours. However, the time needed for access or supply to the controlled area makes for a working day of only six effective hours.

The saw blade experiences upward bending in cases where, from a certain cutting depth, there is material to be cut on only one side of the saw blade. This has not prevented us from cutting the lower core support assembly horizontally into rings. However, the bending of the saw blade must be kept low to ensure a straight cut, in view of subsequent clamping of these rings for vertical cutting. This can be done by first cutting to a depth where there is material on both sides of the saw blade and only then cutting to a greater depth into the work piece.

Core baffle locating pins are the furthest inside the lower core support assembly, i.e. 153mm from the outside diameter of the workpiece. Although cutting them in the radial direction is possible with the equipment, it would imply vibration. Hence, they have to be cut in the tangential direction whilst ensuring a saw blade depth greater than the pin depth of 153mm.

Conclusions of cold test

The mechanical cutting technique using a saw blade works well for complex geometries like the LCSA. However, swarfs collection was found to be a major problem. This was mainly due to the geometry of the workpieces. Hence, most of the testing time was devoted to resolving this problem. Based on the performance of the cutting technique in the cold tests, cutting parameter ranges and operation times have been derived. Apart from the hydraulic pump failure, reliability of the mechanical cutting technique is good.

Table 3 - Suggested values for cutting parameters based on the results of the cold tests

Saw blade		Rotational speed	Peripheral speed	**	Cutting depth /pass	Feed
Diameter	Pitch					
[mm]	[mm]	[RPM]	[m/min]		[mm]	[mm/min]
600	19	6/7	11.5 to 13.4	Entry Exit	25 to 30 25 to 30	25 to 30 15 to 20
600	9.6	6/7	11.5 to 13.4	Entry Exit	20 to 25 No limit	10 to 15 20 to 25
600	5*	7/8	13.4 to 15.3		80	5 to 6
400	19	9/10	11.3 to 12.6	Entry Exit	25 to 30 25 to 30	15 to 20 15

* Core baffle cutting
** For an explanation of the entry and exit side of a cut the reader is referred to Figure 3

Table 4 - Estimated times for set-up and manipulations during mechanical cutting of the lower core support assembly, based upon actual "cold test" times

Configuring and positioning the turn-table and positioning of the lower core support assembly on the turn-table for cutting at a particular level takes 1 day
Turning the turn-table, presenting the LCSA for a new cut and positioning the saw blade and its drive takes - without having to remove the upper supports*: 5 to 10 minutes - having to remove 2 (or 4) upper supports: about ½ (respectively 1) hour
Changing a saw blade takes about 1½ hours
Re-positioning the saw blade for the next pass (retracting radially, returning along the workpiece and positioning at the new radial depth) takes on average 10 minutes
Taking tangential X and Y values takes about ½ hour, unless it can be done during the return. Then it takes about 15 minutes.

*Upper supports are used during the two last cuts at each level to reduce vibration and to prevent movement of the top part of the workpiece which is not clamped on the turn-table.

Table 5 - Example of a calculation of the duration of a full mechanical cut. It is representative of a level IV cut on the lower core support assembly using the cutting plan of Figure 2, the operation time estimates of Table 2 and the values of the cutting parameters of Table 1.

Action		Entry		Exit		Time	#	Total time
		Dist.	Speed	Dist.	Speed			
		mm	mm/min	mm	mm/min	min		min
Positioning of upper support H across the RH tube Ø40/36, reference						5	1	5
Positioning for cut 1						10	1	10
Cut 1 (1 pass)		201	20	171	15	21	1	21
Positioning for cuts 2 to 13						10	12	120
Cuts 2 to 13 (1 pass)		64	20	171	15	15	12	180
Positioning for cut 14						10	1	10
Cut 14 (1 pass)		201	20	171	15	21	1	21
Positioning for cuts 15 to 26						10	12	120
Cuts 15 to 26 (1 pass)		64	20	171	15	15	12	180
Change from Ø400 mm to Ø600 mm milling blade								90
Positioning of 4 upper supports of turn table & positioning for cut 27								60
Cut 27	Pass 1	185	20	122	15	17	1	17
	Positioning for pass 2					10	1	10
	Pass 2	249	20	174	15	24	1	24
Lifting and replacing 2 upper supports of turn table & positioning for cut 28								30
Cut 28	Pass 1	216	20	148	15	21	1	21
	Positioning for pass 2					10	1	10
	Pass 2	275	20	196	15	27	1	27
	Positioning for pass 3					10	1	10
	Pass 3	314	20	65	15	20	1	20
	Positioning for pass 4					10	1	10
	Pass 4	303	20	144	15	25	1	25
	Returning					10	1	10
Total cutting time					minutes			1,031
					hours			17

3.4.2. Tests performed with the Metal Disintegration Machining technique (MDM)

Metal disintegration machining is a cutting technique which had been selected as a back-up technique for circular sawing of difficult or complex geometries, e.g. in the case of cutting the core baffle (Picture 7). Additionally, the technique is to be used for the cutting of tubes - and other relatively thin obstacles - present on the outside of the lower core support assembly or between the core baffle and barrel (Picture 6). The latter obstacles have to be cut by metal disintegration machining to allow the circular saw to enter and cut deeper into the lower core support assembly or the core baffle.

Metal disintegration machining is similar to electro discharge machining. The main difference being that MDM allows contact between workpiece and electrode. The use of a hollow electrode with internal suction is made possible by the oscillation of the electrode. The feed is automatically controlled on the basis of the voltage difference between electrode and workpiece. When contact is detected the electrode retracts and when contact is broken the electrode advances again. The feed is controlled via an electrical motor and a rack and pinion drive. These, together with the electrode are fitted on a table allowing tangential translation of the cutting head. This table fits on the base of the circular saw (Picture 8).

The first "cold" cutting tests were performed in October 1993. Two tubes of Ø40/36 were cut at level IIbis of the lower core support assembly (LCSA) mock-up. This took a total of three hours. An average speed below 0.3 mm/min was found to be too slow.

On the 3rd of November 1993 two similar tubes of Ø40/36 were cut at level IV of the LCSA mock-up. This took five hours. It was tried to adjust the feed control. However, adjustments were done blindly since there was no proper indication on the actual performance of the machining. The average feed was smaller than 0.2 mm/min, which was obviously too slow!

On the 5th of November further cutting tests were done on a different structure, the core baffle. Again, low machining speeds and difficulties in the adjustment of the feed control potentiometers resulted from a lack of operational information. Towards the end of the cut the electrode was broken. Material of the core baffle had entered the suction opening, blocked it partially and then opened it (Picture 9).

It was found necessary to perform additional tests. These were performed in December 1993. The tests were done on another section of the core baffle at level IIbis. A list of the missing operational information was drawn up and the following was found to be absolutely necessary for controlling the process:

- electrode current read-out,
- read-out of the voltage between electrode and workpiece,
- displacement (feed) read-out with a resolution of at least 1mm.

Additional problems were encountered further on in the tests on the core baffle when the electrical motors gear drive kept on breaking. This highlighted the necessity of a motor current management system to protect the gear drive. Although the initial breaking of the gear drive was later traced back to the non-parallellity of the guides and the rack, a current management system is still essential in case of unpredictable sudden stops, e.g. at the end of travel.

During the next testing period, early in January 1994, most of the above modifications were implemented. The test did however end again with a broken electrode. Material of the core baffle had entered the suction opening, blocked it partially and opened the electrode (Pictures 9 and 10). This was a recurring problem !

Four aspects contributed to the problem:

- the position of the axis of oscillation :
  as the electrode wears, the front part comes closer to the axis of oscillation and this reduces the amplitude of the oscillation
- the torsional stiffness of the electrode :
  the amplitude of the oscillation of the electrode is decreased from the foot to the tip of the electrode
- the thickness of the electrode :
  the metal sheets which form the suction channel are only 1.25 mm thick and easily deformable
- the reduced suction of the electrode :
  the suction flow is reduced the moment material enters the opening and welds itself to the inside of the electrode opening.

A new design of electrode has been discussed and will be tried later.

The following preliminary information can be given with respect to cutting parameters:

- Primary current through the electrode:	± 300 A
- Voltage between electrode and workpiece:	± 4 V
- Cutting speeds	0.5 mm/min for tubes Ø40/36
	0.5-2.5 mm/min for core baffle

3.4.3. Vertical mechanical cutting (band saw) (Task B.5.1.)

As described previously [2], the Lower Core Support Assembly will be cut vertically into segments after it was cut horizontally into "slices" by circular sawing. The mechanical cutting technique selected for cutting these rings into segments was the band sawing. After some prototype tests carried out on a small machine for assessing the feasibility of such cutting process in the specific environment [1], the specifications of the band saw machine were drawn up. The machine has to cut vertically the LCSA and the vessel collar as well as the different plates coming from the internals dismantling. The development of this band saw machine is given in the next paragraph.

3.5. Bandsaw machine development for vertical cutting of the LCSA (Task B.5.1.), the reactor collar (Task B.6.) and the plates (Task B.5.2.)

3.5.1. Introduction

Rolls-Royce and Associates have entered into a partnership agreement with SCK•CEN (Studiecentrum voor Kernenergie•Centre d'Etude de l'Energie Nucléaire) to technically co-ordinate the design, manufacture, development and exploitation of an underwater bandsaw. The bandsaw is to be used as part of a range of tools for the dismantling of the BR3 Pressurised Water Reactor at Mol, Belgium. It is envisaged that the bandsaw will be used both in Phase 2 (segmenting of the reactor internals) and the future Phase 3 (segmenting of the reactor pressure vessel).

This paragraph describes progress of the underwater bandsaw project during the period August 1993-December 1993.

3.5.2. Overall design objectives

To design, develop and manufacture an underwater bandsaw capable of cutting the following components into pieces with a "footprint" size less than 500 mm x 500 mm and a thickness less than 500 mm :

a) Lower Core Support Assembly (LCSA) "rings" produced from the horizontal cutting programme of the underwater milling machine;
b) Six circular plates produced from the dismantling and horizontal cutting programmes;
c) Reactor vessel collar.

3.5.3. Status of project at 31.12.93

The bandsaw has been manufactured and assembled. Initial trials, which include tests underwater, will commence during the first week of 1994. Despatch of the bandsaw to Mol for "cold" trials is now programmed for mid January 1994.

3.5.4. Description of the underwater bandsaw

The bandsaw, Figure 6, is manufactured as two main sub-assemblies : support frame and bandsaw head. This has been necessary to allow it to pass through the airlock and into the reactor hall (see picture 11).

Support frame

The support frame is a welded box section fabrication of 304 stainless steel. This is mounted on an adaptor plate secured to one of four work station positions connected to the work-piece turn-table. The adaptor plate has two radial mounting positions to accommodate the range of work-piece sizes placed on the turn-table during Phase 2. A different adaptor plate will be required for Phase 3 work.

Located within the bandsaw frame are two linear rails on to which are mounted the four bearing housing assemblies which carry the bandsaw head. The bearings are of a plain bush design made from a material composed of synthetic fibre impregnated with thermosetting resin. This material has been specifically chosen for its unlubricated underwater performance.

A central leadscrew provides vertical movement for the bandsaw head through an aluminium bronze nut.

Bandsaw head

The bandsaw head is also a welded box section fabrication of 304 stainless steel. It is secured to the four bearing housing assemblies on the support frame. Mounted on the bandsaw head are the sub-assemblies associated with the drive, tensioning and guidance systems of the bandsaw blade.

A 2.4 kW hydraulic motor drives the bandsaw blade through a pinion which meshes with the internal gear ring of the powered wheel. This wheel is of composite construction with a stainless steel tyre fastened to an electroless nickel plated carbon steel hub. The remainder of the bandsaw blade's path is dictated by the positioning of three idler wheels. These wheels are also of composite construction having stainless steel tyres with an interference fit on to hard anodised HE 30 aluminium hubs. Wheel bearings have been chosen to eliminate the potential for release of lubricating oil to the reactor pond. A double lip seal is fitted to the support shaft, while a bearing cap sealed with an "O" ring is fitted over the end of the stub shaft.

Blade tension is achieved by hydraulically adjusting the position of one of the idler wheels located on a linear slide. A constant tension is applied to the blade by an hydraulic cylinder held at a pre-set pressure. Adjustment of the bandsaw blade tension is carried out from the control console by interlocked push buttons which ensure that pressure is maintained during cutting operations.

Two blade guide assemblies are attached to the bandsaw head at the extremities of the throat. The assembly on the drive side can be moved, prior to installing the machine on its work station, to provide an effective throat width of between 400 mm and 500 mm. The stainless steel guides with tungsten carbide wear pads are adjustable to allow for differences in blade thickness and to compensate for wear. To allow for horizontal as well as vertical cutting, the guides can be rotated through 90° by hydraulic cylinders operated from the console (see picture 12). Once again the push button controls are interlocked to ensure no rotation is initiated during cutting operations.

The bandsaw blade is changed by releasing the bandsaw from the adaptor plate and bringing the complete machine to the pond surface.

Vertical and turn-table drives

A 1.5 kW hydraulic motor is used to power the vertical drive gearbox attached to the top of the leadscrew. The worm and wheel gearbox is designed to produce an output speed range equating to a vertical feed rate of between 3 and 30 mm/min. The gearbox casing and end caps are manufactured from hard anodised HE 30 aluminium. The gearbox is protected and internals from ingress of pond water by "O" ring seals and double opposed lip seals fitted to the end cap and shaft respectively. The drive can be readily disengaged to allow the leadscrew to be manually rotated by long reach tooling. This allows a more rapid vertical positioning of the bandsaw head, prior to or immediately after cutting.

Active control of the feed rate is achieved by monitoring the load on the bandsaw blade and using the derived signal to control the speed of the vertical drive motor. Automatic control of feed rate can therefore be achieved to ensure the optimum cutting speed.

Powered drive to the turntable is provided by a 2.4 kW hydraulic motor through a worm and wheel gearbox of similar construction to that of the vertical feed. A quick fit/release mechanism allows a rapid change-over between manual and powered drive systems.

Vertical position transducer

A linear absolute encoder is fitted to give an accurate measure of vertical position and vertical feed rate. This is essentially an aid to the machine operator. It is in addition to the less accurate measure of vertical feed rate provided by transducers fitted to the hydraulic control circuit.

Control Console

The electro-hydraulic control console is positioned at a convenient location within the reactor hall. Hydraulic hoses connect the site power pack to the valve manifold mounted within the base of the console. An electrical umbilical can be connected to the power pack to facilitate remote start/stop operations. Hydraulic hoses also lead from the console to the hydraulic motors and cylinders located on the bandsaw.

The top face of the console contains the operational controls, electronic and pressure gauge read-outs of machine operational parameters, and illuminated indication of machine status.

Manual override of the automatic load sensing is provided. This allows the operator the option of manually controlling the feed rate of the machine by referring to the data displayed on the console.

3.5.5. Design for ease of decontamination

The following design features were adopted to assist in the decontamination of the bandsaw after use :

a) removable bandsaw tyres;
b) removal and containment of the swarf at source;
c) stainless steel components and fabrications produced with a smooth surface finish;
d) aluminium components hard anodised and surfaces sealed with a polymer impregnating process;
e) all welds dressed and passivated;
f) all box section ends sealed and general avoidance of cavities;
g) careful selection of materials for both decontamination and with regard to subsequent disposal.

3.5.6. Cutting considerations

The bandsaw has a vertical cutting capability of 960 mm with a throat width adjustable between 400 mm and 500 mm. These dimensions control how the various components are subsequently cut.

The proposed cutting plan for the LCSA, the "plates" and the collar have been previously reported.

Table 6 - Vertical cutting plan for the LCSA "rings"

Ring identification (see figure 5)	Origin	Shell name	No of vertical cuts	Special segments	Mean width of the segments (mm)	Position of first cut
1	LCSA	Core barrel	8	-	450	Free
2	LCSA	Core baffle	8	-	420	Hot leg axis
3	LCSA	Core barrel	8	-	450	Free
4	LCSA	Core baffle	8	-	410	Hot leg axis
5	LCSA	Core barrel and lower core support barrel	8	-	450	21° from the hot leg axis
6	LCSA	Core baffle	8	-	410	Hot leg axis
7	LCSA	Lower core support barrel	8	-	410	Free
8	LCSA	Lower core support barrel	9	*Hot leg collector drain pipe *Radial guide spacers	< 500	Specific
9	LCSA	Lower core support barrel	9	*Reinforcing ribs *Pipes	< 500	+25° from the hot leg axis
10	LCSA	Lower core support barrel	10	*Reinforcing ribs *Pipes	< 500	Specific
11	Collar		11	-	< 500	Specific
Total			95	-	< 500	

Table 7 - Summary of plates geometries

No	Name	Part of	OD (mm)	Thickness (mm)	Overall thickness (mm)
1	Rod shroud hold down plate	-	1330	35	-
2	Rod shroud support plate	Collar and instrumentation basket	1405	37	215 (with the reinforcing ribs)
3	Intermediate instrumentation basket plate	Instrumentation basket	904	35	-
4	Lower instrumentation basket plate	Instrumentation basket	904	35	380 (including the top hats)
5	Spray box	UCSA	1066	-	75
5.1.	Upper core support plate	UCSA	1066	35	-
5.2.	Spray distribution box	UCSA	1066	40	-
6	Lower core support assembly	LCSA	1200	35	-

3.6. Handling and supporting equipment

3.6.1. The turn-table

As reported previously, a supporting table or "turn-table" was designed, manufactured, delivered and cold tested during the period covered by the report.

The turn-table involves four successive steps :

Step 1 : includes the base structure and the bearing, foreseen to receive directly the spray box, the instrumentation basket and the remote controlled telemanipulator. It is illustrated on picture 03. Its delivery took place in August 1993. The purposes of the cold tests were to verify the calibration and the relationship between the torque applied and the clamping forces produced, and to train the operators.

During these cold tests, the good lubrification quality of the water was demonstrated. The torque needed to clamp the workpiece under water with the right clamping force was smaller (25 %) than the torque applied out of the water. During this cold test, the necessary tools for the clamps and for the rotation of the turn-table were also checked.

Step 2 : foreseen to clamp the workpieces (LCSA) during the horizontal cutting, it includes step 1 and an additional modular plate, the so-called "base plate", with its associated clamping device. It was delivered at the beginning of July. Its cold test was performed at the same time as the cold test of the circular saw. Picture 13 illustrates the configuration of the additional modular plate foreseen for horizontal cut no.1 of the LCSA and picture 14 shows the turn-table in step 2 configuration supporting the workpiece during the circular sawing test. To avoid any movement of the workpiece during the cutting, the foreseen clamping force values are : 2 tons by clamp for shells and 0.5 ton by clamp for the core baffle. Some trials were executed during these cold test to demonstrate the necessity of these force values : with smaller force values, the workpiece vibrated during performance.

Step 3 : foreseen to clamp the workpiece produced by the horizontal cutting of the LCSA as well as the collar during the vertical cutting, it includes step 1, the modular base plate of step 2 and specific clamping devices.

The number of the clamping devices used depends on the ring size.

Picture 15 illustrates one clamping device unit. Picture 16 shows step 3 of the turn-table with a workpiece clamped. To manage the different sizes, forms and some protrusions of the workpieces and to minimize the number of the needed parts, it was necessary to :

- install each clamping device unit on an adaptor plate between the clamping device unit and the base plate;
- make the support of the upper locking system modular.

Taking into account the 11 rings to be cut, the number of modular parts is greater than 275 different items and the number of the bolts is about 750. More than 2 000 actions of bolt screwing and unscrewing will be needed for the whole cutting campaign. Picture 17 illustrates a part of the modular pieces needed to clamp the 11 rings.

Step 4 : foreseen to clamp the 6 plates to be dismantled, it includes the base plate of step 2 with its clamping device and 2 modular plates to position and to fasten the clamping device on the base plate (see figure 7).

The two adaptor plates were designed. The fabrication and assembly will be carried out early in 1994.

3.6.2. Handling equipment

Linear gantry crane across the reactor pool

As explained in the previous report, a linear gantry crane with an electrical hoist of 10 tons was needed for the handling operations.

This equipment has been manufactured, installed and checked by an official organism during this period.

The erection time necessary to install it was less than one week and the operations were executed by three persons with a total distributed dose of less than 0.24 mSv.

The gantry crane is now ready for the loading of the turn-table, the handling of the internals and other works.

Handling of the cut pieces

For the remote handling of the rings and plates or flanges, two specific gripping devices were manufactured, delivered and tested during this period. Picture 18 illustrates the specific gripping device for the rings which were used during the cold test to handle the rings produced by the horizontal cutting of the LCSA mock-up. These gripping tools can be easily locked and unlocked remotely.

One of the three gripping clamps for the handling of plates or flanges is illustrated on picture 19.

3.7. Summary results of the hot performance and the cold tests

The first technical operations on the actual pieces have been carried out without major problems and with less dose distributed to the workers than expected. The dismantling of the control rods drive shafts was carried out "hands-on" due to the very low dose rate generated by these pieces. One of the results of this operation concerned also the generated waste : most of them can be considered as "standard" low level waste (< 2 mSv/h contact dose rate), the remaining ones being recategorized to this same type of waste by quite simple decontamination operation. This result has important consequences for the estimated amount of medium level or high level waste to be expected from the dismantling of the upper internals. This results also from the chemical decontamination of the primary loop performed in 1991 in the framework of phase 1 of the pilot project.

The second operation (the removal of the rod shroud support plate) using the impact unbolting and the EDM cutting was carried out with success. The problem free execution was probably due to the intensive cold testing of the operation on real scale mock-ups, where most of the arising problems were solved. This training effect was also visible during the actual execution, where similar operations were carried out with decreasing distributed dose.

The EDM showed also its capability to carry out difficult operations on complex pieces (e.g. the oblique cutting of unaccessible bolts) as well as unforeseen "surgical" operations like the cutting of the instrumentation tube.

The mean results of the EDM cutting can be summarized as follows : for desolidarization of a bolt M12 in a plate with a thickness of 37 mm, the following mean duration and waste generation were noticed :

Plate thickness (mm)	Type of cut	Cutting depth (mm)	Electrode OD/ID (mm)	Cutting duration (h)	Overall cutting operation duration (man-day)	Number of generated filter cartridges
37	Vertical	42.2	22/16	1.14	78	20
37	Oblique	65.6	22/16	2.37		

For the remaining operations of phase 2 of this pilot project, most of the dismantling techniques were cold tested on partial full scale mock-ups. This concerned the hydraulic jaw cutter for cutting remotely pipes and tubes, the telemanipulator for positioning the jaw cutter, the horizontal cutting of revolution assembly (the LCSA) using the circular sawing and the MDM cutting and finally the so-called turn-table for supporting and clamping the workpieces and the dismantling tools. The hydraulic jaw cutter and the circular sawing showed good results and were considered as being ready for the dismantling of the actual workpieces. The remote telemanipulator and the MDM cutting process require further actions to be able to use them on the actual workpieces. Indeed, as it was shown for the removal of the rod shroud support plate, actual operations can be carried out with very few problems and minimizing the distributed dose to the operators only when all the problems encountered during cold testing have been resolved before entering the controlled area.

Finally, the dose estimation and follow-up, as well as their comparison are used to better model the future operations and then be better able to propose radioprotection optimization actions and to better estimate their influence on the distributed dose.

3.8. Optimization of the pool space occupation and removal of the collar support structure and shielding

The support structure of the collar together with its lead shielding remained on the bottom of the pool.

Their unloading and dismantling were carried out to create more space in the pool for the future operations and for intermediate storage between horizontal and vertical cutting of the LCSA.

Indeed, during the horizontal cuts of the LCSA, it is necessary to have three free spaces :

- one for the storage of the cut rings
- one for the remaining part of the LCSA
- one above the turn-table, for unloading the cut ring.

These two supporting and shielding equipments were stored in the pool for 30 years and during each refuelling, they supported the collar with its instrumentation basket. After each refuelling and closing of the reactor head, the lead shielding was emptied by means of a pump and the internal contamination was fixed by paint. So the level of its internal contamination is very high. The lead shielding is an open cylindrical tank composed of a lower circular plate (50 mm thickness) and three rings. These rings in their turn are composed of two shells with lead between them. Each shell was welded to the other one, the last one being welded to the circular plate (see figure 8). The total weight is about 6.3 tons.

The support structure comprises a circular base plate, three columns and a "crown" at the top. The three columns were reinforced by cylindrical plates. All material is painted carbon steel.

The evacuation of the lead shielding was carried out as follows.
First, a specific lifting device had to be designed and manufactured (see figure 8). Second, the water level in the pool had to be lowered up to the top level of the lead shielding. This enabled to empty the lead shielding by means of a pump and to refill it repeatedly with clean water. After several cycles and a final cleaning by high pressure water jet, the contamination was measured and the installation of the specific lifting device was allowed. The lead shielding could be withdrawn from the pool (see picture 20) The measurement of the contamination before and after cleaning is shown on figure 9.
Taking into account the high contamination level of the pieces, one decided to dismantle the lead shielding inside a ventilated tent. This dismantling was carried out by grinding the welds between the rings. To avoid any further dispersion of the contamination, the lead shielding has been painted afterwards.
Each cut ring was packed in a plastic bag and evacuated from the reactor building to the Shipping Area (auxiliary building) where it is stored in a rigid box to avoid any mechanical damage of the plastic cover.

The dismantling and evacuation of the support structure of the collar was carried out following the same procedure. First, horizontal cutting by grinding in two pieces; second, vertical cutting of the upper part in three pieces. The lower part (packaged) remained in the reactor building. The other three pieces were evacuated from the reactor building to the Shipping Area in the same way as the pieces of the lead shielding.

All information regarding the ALARA approach and the distributed doses are given in paragraph 3.9.

The contamination level of the air in the tent was the following :

- βγ contamination : 78.3 Bq/m^3
- α contamination : equal to the existing background.

The future operation for this equipment will be the decontamination of the walls by means of sandblasting with the aim of reaching the free release level.

After the work was carried out, the tent was cleaned, removed and stored for future works.

3.9. Radioprotection optimization

During the period covered by this report two main dismantling activities have been the object of special follow-up work from the radiological point of view. The aim of this follow-up is not only to record doses and to relate them to their corresponding tasks but also to validate the method used for predicting them.

Moreover, recording and storing doses received by workers will be useful for planning future jobs. It will allow us to assess which parameters or procedures should be changed to reduce doses according to the ALARA requirements.

First we have made a dose estimate for the removal of the support of the collar and its lead shielding (see paragraph 3.8.). This dose estimate was made as described in the last progress report [1].

We obtained a collective dose value and a collective dose distribution for each task that would be compared to those obtained during the execution of the various operations.

To be able to assess the validity of the dose estimate, we ensured a close follow-up of the work, so that the doses measured could be approximately related to the tasks carried out.

Table 8 shows the results of the comparison made between estimated and observed doses for each task into which the work has been divided.

As far as the removal of the lead shielding is concerned, the overestimated dose values refer to the ring cutting tasks and, particularly, to the resulting packing work. The underestimated dose values concerning the removal of the support are related to certain tasks involving the dismounting and packing the upper part of the support. Figure 10 shows graphically all ratios between estimated and observed doses.

The average ratio between predicted and actual dose values is 1.6 which is a satisfactory result if we consider the degree of uncertainties existing before carrying out the work.

The second dismantling activity, for which we have made a follow-up of the doses, was the removal of the so-called "rod shroud tubes support plate" (RSSP) (see also chapter 3.2.1.).

The operations sequence is shown in figure 11. As it can be seen the EDM tasks had a "symmetric" character, i.e. in almost each pair of tasks, one of them was carried out in a position on the plate that was symmetric to that of the other task. The result of this "symmetry" was a decreased amount of the distributed dose as, carrying out the first task, this acted as training to execute its symmetric task; this is called the "learning effect". It allowed us to carry out the latter faster than the former. It applies for instance to EDM1 (0.204 mSv) and EDM2 (0.131 mSv) tasks, EDM4 (0.175 mSv) and EDM3 (0.124 mSv) etc.

Here the dose estimates and the optimization exercise had been made well in advance [2]. Table 9 shows the comparison between the dose estimates and the observed doses. The ratio between predicted and observed dose was about 3.4.

Overestimation of the dose intake is due to :

- an overestimated number of workers needed to carry out the work;
- most of the positioning with the X-Y table has been carried out from a higher level than what had been foreseen (platform at level 7 m instead of platform at level 2 m);
- an overestimation of doses during filter replacement tasks.

Besides the above and according to the ALARA principle, several lead shields (Pb blankets) were set up around the service platform (level 2 m). This has contributed to a reduced radiation level in the working area next to the reactor vessel (120 μSv/h instead of the predicted 160 μSv/h).

For the individual dose distribution, table 10 shows that the values have never exceeded 2 mSv (monthly SCK•CEN limits). Although the highest value has been 1.9 mSv, this has only been the case for one person, the rest of the values remaining lower.

Finally, the next planned activities will be the continuation of the follow-up of the dismantling operations having the workers' radiation exposure reduction as the main objective. We believe that this work will be useful for achieving a good radiation optimization during further dismantling operations.

Table 8 - Comparison between estimated and observed doses

TASKS	ESTIMATED DOSES (man-mSv)	OBSERVED DOSES (mans-mSv)
Removal of the Pb shielding		
*Transfer	0.56	0.35
*Cutting	1.02	0.34
*Packing	0.24	0.06
Removal of the support		
*Transfer	0.17	0.10
*Dismounting	0.07	0.24
*Cutting	0.14	0.17
*Packing	0.02	0.07
Evacuation	0.4	0.33
TOTAL	**2.62**	**1.66**

Table 9 - Collective dose per task

TASKS	ESTIMATED DOSES (man-mSv)	OBSERVED DOSES (man-mSv)
Preliminary work	1.46	0.98
Unbolting	16.5	0.44
EDM bolt cutting		2.93
Unbolting of the plate		0.27
Plate removal		0.35
Others		0.38
TOTAL	**17.96**	**5.35**

Table 10 - Individual doses distribution

DOSE RANGE	% OF THE PERSONNEL INVOLVED
< 0.1 mSv	59
0.1 - 0.5 mSv	23
0.5 - 2 mSv	18

4. PROGRAMME OF THE WORK FOR THE NEXT PERIOD

The work in the next period will mainly cover the dismantling performance on the actual pieces. It will thus comprise :

- the achievement of the desolidarization between the vessel collar and its instrumentation basket;
- the unloading of the upper internals and the collar, separately;
- the dismantling of the instrumentation basket;
- the dismantling of the Upper Core Support Assembly (so-called "spray box);
- the unloading of the Lower Core Support Assembly (LCSA);
- the horizontal cutting of the LCSA into slices;
- the vertical cutting of the LCSA slices;
- the vertical cutting of the vessel collar.

The overall time schedule for the whole project is given hereafter.

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- the horizontal cutting of the LCSA into slices;
- the vertical cutting of the LCSA slices;
- the vertical cutting of the vessel collar.

The overall time schedule for the whole project is given hereafter.

Programme Item N°	Year and Quarter											
	1991		1992				1993				1994	
	III	IV	I	II	III	IV	I	II	III	IV	I	II
(Phase 1 : B.1. - B.3.)			WASTE EVACUATION									
Phase 2 :												
B.3.												
B.4.												
B.4.1.												
B.4.2.												
B.4.3.												
B.5.												
B.5.1.												
B.5.2.												
B.6.												

Legend :

Subdivision of Phase 2 into main tasks :

B.1.3.-B.3.	Phase 1 tasks
B.3.	Extension of this Phase 1 task (generation of specific data) to the new tasks of Phase 2
B.4.	Disassembling the main reactor internals
B.4.1.	Lower Core Support Assembly
B.4.2.	Upper Core Support Assembly
B.4.3.	Reactor Vessel Collar and Instrumentation Basket
B.5.	Segmenting of Stainless Steel reactor internals subassemblies
B.5.1.	Annular cylindrical geometries (3D geometries)
B.5.2.	Plates or grids (2D geometries)
B.6.	Segmenting of a thick Carbon Steel ring (Reactor Vessel Collar)

5. LIST OF FIGURES

6. LIST OF PICTURES

Picture 19 One of the three gripping clamps for the handling of plates or flanges.

Picture 20 Withdrawal from the lead shield out of the refuelling canal.

Waste and Clean-up Division
Dpt. BR3 Decommissioning

RPV AND INTERNALS DISMANTLING PROJECT

(BR3 - EWN - KRB-A)

Final report

Research Contract No. FI4D-CT95-0001

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EUR XXXXX EN
August 2000

This report is a common report SCK•CEN (Belgium), KRB-A (Germany) and EWN (Germany)

Table of contents

1. ABSTRACT

As follow up of the pilot decommissioning projects carried out during the 4th framework programme of the Commission, this contract involved three major reactor decommissioning projects, in which the dismantling of the most radioactive parts, i.e. the reactor vessel and internals, were concerned.

These three projects involved three of the principal nuclear reactor types to be found on the European Continent: the Pressurized Water Reactor (PWR), the Boiling Water Reactor (BWR) and the Russian type pressurized water reactors (VVER).

The three projects were:
- the BR3 reactor, PWR situated in Mol, Belgium;
- the KRB-A reactor, BWR situated in Gundremmingen, Germany;
- the EWN reactors, 4 VVERs situated in Greifswald, Germany.

The main aim of the project was the testing, comparison and execution of the dismantling of highly activated reactor pressure vessel (RPV) and internals. Most of the internals of the BR3 and KRB-A reactors were already dismantled during the previous framework programme.

The principal objectives of this project were thus:
- the remote dismantling of the BR3 RPV;
- the remote dismantling of the KRB-A RPV;
- the preparation, testing and model dismantling of the EWN reactors internals and RPVs.

Due to unforeseen events during the execution of the projects, all the planned activities were delayed, and an extension of the contract duration was requested. Nevertheless, even so, some of the project objectives were not reached completely. But the main comparison, study and technology testing, were carried out and will be reported in the present document.

This is thus the report of a European première concerning the complete dismantling and size reduction of reactor pressure vessels and internals of the most widely spread reactor types.

2. GENERAL PRESENTATION

BR3

The BR3 reactor (Belgian Reactor nr.3) was the first PWR ordered and connected to the grid in Western Europe. It was ordered to the Westinghouse company (US) in 1956 and started operation in October 1962. The reactor had a thermal power generation of 40.9 MWth, a gross electricity output of 11.5 MWe for a net power output of 10.5 MWe.

The BR3 was mainly used for training commercial reactor operators and for testing advanced fuels (high burnup, burnable poison, MOX fuels) in full PWR conditions.

During its lifetime, starting in 1962 and ending on 30th June 1987, the BR3 has produced 964.3 GWh of electricity in 11 operating campaigns and has undergone three main operations:

- In 1964 the reactor internals were removed and exchanged by new ones for carrying out an experiment called "Vulcain". This experiment involving a mixture of heavy and light water required also some changes to the auxiliary loops in order to control the mixture composition and to recover the heavy water.
- In 1975 the primary loop was decontaminated by a chemical process called Turco.
- Finally, in 1984, one carried out a wet annealing of the pressure vessel to decrease the neutron induced embrittlement of the RPV material and thus to allow further operation of the plant.

On June 10th, 1987 at 24.00 hrs, the reactor was definitely shut down.

The BR3 decommissioning was selected in 1989 by the European Commission as one of the four pilot projects in the framework of its five year RTD programme on decommissioning nuclear installations. The first contract was signed in October 1989 with an effective commencement date on 1st October 1989.

The operations were divided into two main phases, as requested by the Commission.

The **first phase** comprised mainly the carrying out of a pre-dismantling full system decontamination of the primary loop, the selection and testing of cutting techniques for the dismantlement of highly radioactive internals and then the actual dismantling of a first reactor internal, the thermal shield.

The **second phase** involved the dismantling of all the remaining reactor internals.

At the end of the contract, an extension was signed which involved the dismantling of a first set of internal pieces, unloaded in 1964, and the comparison of this operation (concerning pieces with a cooling down period of 30 years) with the preceding one, done almost immediately after shut down. This phase was called the **phase 2bis** of the pilot project.

The next phase of the project concerns the dismantling of the reactor pressure vessel (RPV). The pressure vessel is a 28 ton carbon steel (ASME SA 302 gr.B Ni Modified) forged piece clad with stainless steel (ASME SA 240 gr.S).
Different strategies for the RPV dismantling were studied and led to the selection of underwater cutting, the RPV being removed from its pit into the refuelling pool.
The pressure vessel had then to be decoupled from its primary loop and unfastened. The refuelling pool leak tightness had to be reinstalled. This led to the RPV removal, carried out in August 1999.
The real dismantling of the RPV could then proceed.
The thermal insulation and insulation shroud were completely removed in December 1999 and from January 2000 till June 2000, the main RPV shell has been cut into pieces fitting in 400 liter drums (Belgian standard waste package).

EWN

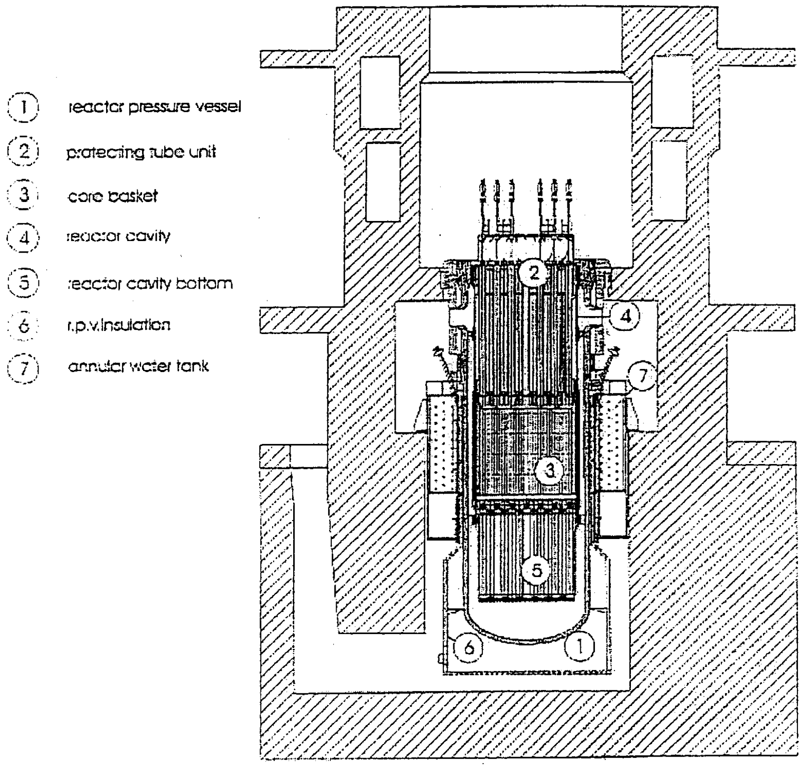
At the Greifswald site there are 8 units of the Russian pressurised water reactor type WWER 440 (**see figure 1**). After the reunification of Germany, the units 1 to 4 were switched off and the trial operation of unit 5 and the construction works on the units 6 to 8 were stopped. For economical reasons, it was decided to immediately remove and not to enclose the facilities.

Due to the high activation level of the components of reactors 1 to 4, it was decided to use mainly remote handling technique for their dismantling. For the dismantling and cutting of the reactors 1 to 4, the concept first foresees a model dismantling to test the transport, cutting and packaging equipment to perform the later dismantling of the activated reactors 1 to 4 safely and optimally.

The not contaminated equipment of the units 7 and 8 will be transported into the steam generator room of unit 5 and will be cut there in the frame of the model dismantling. The tested tools and equipment will then be used for the dismantling in units 1 to 4.

For reactor unit 5, the cutting of the reactor components is not planned due to the short operation time and the low-activation level. The individual components will be transported as one part to the interim storage on the site and after a decay storage of 40 to 70 years, they will be cut without remote techniques.

After the dismantling of the operational equipment, the preparations for the installation of the cutting equipment in steam generator room unit 5 were executed from June 1997 until March 1998. Approximately 780 t have been removed from the wall and floor areas to have space for the installation of the cutting equipment. From March 1998 up to September 1999, the main equipment was installed, and since December 1998 the systems have been under commissioning. In October 1999, the reactor pressure vessel was put into the dry cutting place and thus, model dismantling could start in unit 5. In comparison to the original time schedule, this meant a delay of approx. 10 months for the start of model dismantling.



reactor components of the units 1 - 4 EWN

Figure 1

During model dismantling, several technical problems occurred, especially with the band saws and video technique which strongly affected or even stopped the tests. For these reasons, only few data are available for the evaluation of the cutting procedures and for the estimation of the time needed as well as for the radiological exposure to be expected for units 1 to 4.

Resulting from the time delay for model dismantling in unit 5 and from the knowledge that for the installation and commissioning in unit 2 a longer time has to be estimated, the start of active dismantling in unit 2 will be delayed for approximately 1.5 years.

KRB-A

The nuclear power station Gundremmingen unit A (KRB A) is located in Bavaria at the river Danube between Stuttgart and Munich. It was under construction in 1962-66 as the first commercial nuclear power plant in the Federal Republic of Germany (FRG) with an electrical output of 250 MW. (**Figure 2)**

KRB A was operated from 1966 until January 1977. The plants average availability was 75 %. It was shut down in January 1977 due to a short circuit in the grid which caused substantial damage to the plant. KRB A determined that necessary repairs and backfitting measures could not be justified from an economic point of view. In January 1980, the shareholders RWE-Energie AG (75 %) and Bayernwerk AG (25 %) decided to decommission the plant. Since 1984 two modern boiling water reactors (BWR's) are in operation on the site with an electric output of 1344 MW each.

The planning phase of the decommissioning of unit A began in 1980, actual dismantling work started in 1983. It is foreseen to decommission the internals of the plant and the reactor building. The other buildings will be kept as a workshop for the two running BWR´s.

KRB A was involved in seven RTD contracts with the dismantling work during the foregoing EC programmes. It is a European enterprise according to the Euratom Treaty.

The dismantling project was separated into three phases. The dismantling work in the turbine hall contained about 4500 tons of material with rather low contamination, up to 1000 Bq/cm^2. It turned out, that the total collective dose in this first phase was not more than 1 Sv.
In Phase II the rather high contaminated primary water affected systems within the reactor building had to be removed.

3. EXECUTION OF THE PROJECTS AND RESULTS

B.1. Project co-ordination

This project, grouping in fact three major D&D European projects, was a collection of important data and technical expertise and involved exchange of information about the technologies used, the problems encountered, the solutions brought and the return of experience.

In order to co-ordinate the project, some meetings were specially organized at the different projects locations, but it appeared that a more efficient, cheaper and regular information exchange system was needed.

Therefore, a short monthly internal progress reporting was set up between the three partners, allowing to keeping the knowledge of the progress in each project up to date at every moment. Moreover, this exchange improved the information exchange when important similar problems were encountered in the projects, like e.g. the water pool turbidity, at KRB-A and BR3.

The co-ordination also took care of the contractual document publication (progress reports and Annual summary Progress Reports or APRs).

In order to limit the costs of travelling and subsistence, most of the internal co-ordinating work for this project was carried out during the foreseen EC co-ordinators meeting where the different partners could attend. Some specific meetings (as mentioned in the table below) were also held when needed.

The meetings are summarized in the following table.

Table 1: Co-ordination meetings of the three projects

Meeting date	Location	Meeting type	Organized by
03-04/06/1996	Brussels	EC co-ordinator (DWG)	EC
13/11/1996	Gundremmingen	Internal	KRB/SCK•CEN
18/11/1996	Brussels	EC co-ordinator (DWG)	EC
29-30/04/1997	Culham	EC co-ordinator (DWG)	UKAEA/EC
03/11/1997	Hannover	EC Co-ordinator (DWG)	UWTH/EC
26/02/1998	Brussels	Preparation	EC/SCK•CEN
19/03/1998	Avignon	EC co-ordinator (DWG)	CEA/EC
19-19/02/1999	Brussels	EC co-ordinator (DWG)	EC

As foreseen by the contract, periodical progress reports were issued. These progress reports were distributed to the EC and among the different FP4 projects on Decommissioning.

B.2. Selection and testing of techniques – General part

B.2.1. Selection for cutting in air or under water

BR3

Based on the experience gained during the previous phases of the pilot project, the BR3 team decided from the early beginning to promote mechanical cutting. For the moment, the band saw is worldwide used and BR3 is proud having been the first one to try this technique remotely and under water.

A first detailed study compared the complete dismantling of the RPV in air or under water (as for the internals in the previous project). For this comparison of dry and wet cutting, the study focused on the following areas:

- technical feasibility;
- the radiation protection and safety of the operators, including the case of equipment failure;
- the shielding needs to cope with the radioprotection requirements.

As summary, it can be said that the collection and filtration of swarfs when cutting in air represent a difficult problem to solve. The PWR plant configuration with a refuelling pool above the RPV (fit with a purification loop) give an important opportunity to have a good shielding (water). This media gives also a good accessibility to the whole underwater workshop, built on the bottom of the pool. The RPV being an active part of the refuelling pool, its underwater removal implies modification to guarantee the water tightness of the pool. The RPV being surrounded by an annular Neutron Shield Tank, the vessel can be submerged (refuelling pool extended to the NST inner wall). Only the three penetrations for the primary pipings have to be closed safely.
So, mainly for technical and safety reasons but also for economical reasons, the underwater dismantling was selected.

A second study was then started to analyse in details two different approaches: the in-situ dismantling where the RPV remains in place while being cut into rings, and the one piece removal, where the vessel is removed in one piece into the refuelling pool, where it will be segmented into pieces ready for packaging.
The advantage of the latter is the accessibility of the RPV and its insulation shroud from the outside, giving the possibility to reuse the dismantling tools and equipment designed for the internals dismantling. Moreover, this approach simplifies greatly the dismantling of the RPV insulation shroud situated about 100 mm outside the vessel wall.

Arrangement of cutting places

For the cutting of the activated reactor components, one cutting place is foreseen for each of two units (**see figure** 3). For units 1 and 2, this cutting place will be arranged in the steam generator room of unit 2 and for units 3 and 4 in the steam generator room of unit 4. For each dismantling area, one dry cutting place is foreseen for the cutting of the reactor pressure vessels and the lower activated parts of the reactor cavities and protecting tube systems. For the cutting of the core basket, cavity bottoms and the higher activated areas of the reactor cavities and protecting tube systems, a wet cutting place will be installed.

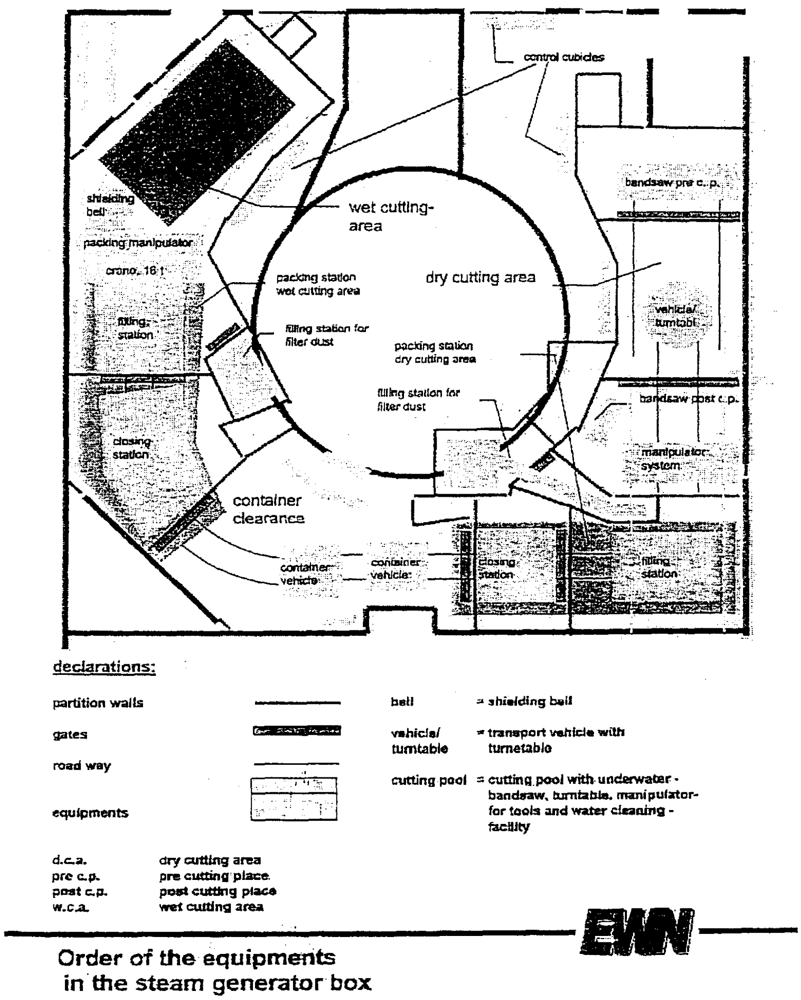


Figure 3

The two cutting caissons as well as the packing station are implanted below openings to the reactor hall, so that the loading and unloading of these areas can be performed without problems with the overhead crane in the reactor hall.

Dry cutting caisson

The dry cutting station is divided in a pre- and post cutting area divided by a shielding wall. Through a vertical gate in the shielding wall, it is possible to move a transport vehicle between the cutting areas. The dry cutting caisson is equipped with an under-pressure control system.

The component to be cut is placed in the pre-cutting area on a turntable. The turntable is mounted on the transport. The component is fixed with a wire hoist. In the pre-cutting area, the horizontal cuts are performed. With the transport vehicle, the piece is subsequently transported to the post-cutting area. There, the vertical cutting takes place into pieces suitable for packing and later final storage. The normal cutting tools are band saws. In the framework of the testing programme also autogenous and plasma cutting will be tested.

In the packing station which is directly connected to the caisson, the containers are loaded and covered with a manipulator. During filling, the container is equipped with a protective sleeve in order to avoid exterior contamination.

Wet cutting caisson

This caisson consists basically of a cutting pool with cutting devices, different transport and handling devices, water purification system, emptying and filling connections and a ventilation system.

The caisson is separately ventilated and kept at under-pressure. Neighbouring rooms are not radiologically influenced.

The component to be cut is placed in the caisson on a turntable. The cutting is normally performed with band saw, horizontal as well as vertical. Optionally, also plasma- and CAMC-cutting devices are foreseen. For the handling, two manipulators are foreseen. One is used for the handling of the cutting tool and the other one for the handling and transport of the pieces, which are placed in wiremesh baskets. The loaded baskets are transported to the packing station with an overhead crane and placed in a container. Subsequently, the container is automatically closed.

Cutting techniques (wet and dry)

Mechanical techniques; dry cutting area

For the pre- and post cutting stations, band saws are foreseen. With this equipment, the pressure vessel as well as the upper part of the protecting tube system and reactor cavity will be cut horizontally into rings (pre-cutting station) and following cut vertically to segments (post cutting station).

Table 2: Comparison band saw pre-cutting and post cutting

Main design features of band saw	Band saw pre-cutting	Band saw post-cutting
Cutting speed (adjustable)	15-80 m/min	15-80 m/min
Saw drive (continuously adjustable)	1-170 mm/min	1-170 mm/min
Maximum cutting length	ca. 3000 mm	1500 mm
Maximum cutting depth	ca. 500 mm	ca. 1100 mm
Total weight	ca. 20 Mg	ca. 10 Mg

On the post cutting place, 2 manipulators with a lifting capacity of 100 kg for the handling of tools and a manipulator with a lifting capacity of 500 kg for the handling of the cut reactor parts are arranged (**see figure 4**).

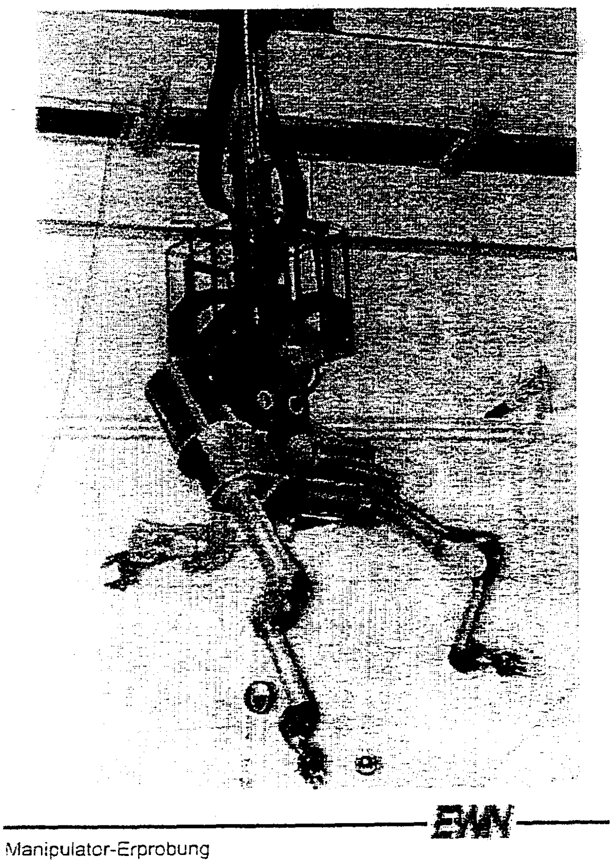


Figure 4: 2 manipulators on the post cutting place

Mechanical techniques; wet cutting area

- underwater band saw;
- underwater abrasive and cutting-off machines;
- underwater shears;
- underwater milling cutter;
- other underwater saws.

A high power band saw (vertical saw) is used as underwater band saw. It consists of a non-distortional upright in cluster-type design and a rigid non-distortional and vibration-free saw frame. The band exchange is performed from a platform of the cutting pond. It is possible to make horizontal and vertical cuts with the band saw, since the sawing blade can be turned.

The main technical data are:

– cutting speed	ca. 15-60 m/min
– saw drive (continuously adjustable)	1-170 mm/min
– cutting depth	ca. 450 mm
– cutting height	1000 mm
– total weight	ca. 30 Mg

Furthermore, there are a manipulator used as tool carrier with a lifting capacity of 100 kg and a packaging manipulator with a lifting capacity of 300 kg on the wet cutting place.

The cutting caisson has the following inner dimensions:

– length	11 000 mm
– width	4 500 mm
– height	5 500 mm (unit 5), 6 600 mm (units 2/4)

Thermal cutting techniques

A thermal cutting device on a band saw frame is mounted at the dry cutting caisson of the pre-cutting place. The positioning of the cutting device to the reactor components is regulated by the feed range of the band saw.

The thermal cutting procedure at the post cutting place is performed by means of the packaging manipulator as a tool support system.

Reactor components of austenitic material will be cut by the plasma cutting device. Components of ferritic material will be cut by autogenous flame cutting.

For the underwater application in the cutting pond, contact-arc-metal-cutting (CAMC) and underwater plasma cutting are intended to be used as thermal cutting procedures.

CAMC is a procedure for thermal cutting of the metal materials by repeated short-circuit high-current arc generated by contact of the electrode and work piece. The melted material is rinsed out of the trajectory by means of a water flushing integrated in the electrode.

Casks

The test of loading, transport and handling of the casks forms a priority for model dismantling.

The concept foresees to pack the dismantled reactor components into transport and interim storage casks. The components will be stored for decay storage of at least 40 years in the newly built Interim Storage North (ISN) on the EWN site.

For the storage in the ISN, the dose rates of the casks have to be below the following limit value: in 2 m distance 0.1 mSv/h for square packages.
Furthermore, a licence for the transport of the casks on road and rail will be needed.

For the storage of the dismantled reactor components (reactor pressure vessel, core basket, reactor cavity and cavity bottom, protecting tube unit and annular water tank), steel and concrete container are foreseen. The dismantling of the absorber and shielding assemblies is not planned. They will be loaded and finally stored in CASTOR casks.

The steel containers are intended to be used for the packaging of the reactor components with a higher activity (reference value for the classification is an activity of 10^{17} Bq/g).

The outer dimensions are:
- length 2000 mm
- width 1600 mm
- height 1450 mm.

The wall thickness is 180 mm or 210 mm.

Corresponding to the activity inventory to be stored, additional inner liners will be used. For the side walls and the cover, 30 mm steel, and for the container bottom max. 20 mm lead can be used.

For the less activated components, concrete container will be used for interim storage. The outer dimensions are equivalent to those of the steel containers, the wall thickness is 200 mm. Heavy concrete with a density of 3.5 g/cm³ will be used as basic material.

The highest activated residues (core basket, reactor cavity, parts of the protecting tube unit) will be provided additionally to the storage in the steel container with a secondary shielding. The activity of the residues must be higher than 10^8 Bq/g. The outer dimensions of the secondary shieldings are:

- length 2660 mm
- width 2260 mm
- height 2060 mm.

The wall thickness is 280 mm.

The secondary shieldings consist of a steel skeleton cover with backfill material of normal concrete with a density of 2.25 g/cm³. A licence for the transport on road and rail is not foreseen.

2 prototypes of containers are foreseen to be tested in the frame of the model dismantling. According to the results the choice will be made and the serial production will be initiated.

KRB-A

For the dismantling of activated components, i.e. the RPV and the internal components, various cutting tools can be used, because mechanical and thermal cutting techniques are authorized at KRB A in principle.
Suitable mechanical cutting techniques are all ordinary milling and sawing tools. Thermal cutting techniques, usually applied at KRB A, are oxy-acetylene cutting, plasma arc cutting and CAMC (Contact Arc Metal Cutting).

The structures of the internal components are not regular, but in the most cases very complex. Therefore, using a variety of tools for dismantling has been proven to be sensible. Mechanical and thermal tools are used in combination, depending on the special cutting task. The main reasons for selecting a tool in this field of dismantling are not of radiological but of technical character, because all work is done remote controlled and under water.

In case of cutting in air, the main emphasis for selecting an appropriate tool is shifted to the reasons of radioprotection. On the one hand, the radiation of the component to be cut must be regarded, on the other hand, the spreading of radioactive aerosols must be taken into account. Basically, mechanical cutting systems cause less aerosol production than thermal procedures. But thermal techniques allow higher cutting speeds and often they are more flexible in view of the non-linear cutting track.

For the RPV the decision for the implementation of cutting techniques in air or under water is depending mainly on radiological aspects. In this context, it is useful to investigate each RPV section with the specific activation and contamination separately. Technical problems in principle are not expected, as the RPV has a regular shape, so that no tool must be excluded from this point of view. Nevertheless, the chosen dismantling technology should fulfil special requirements: a low aerosol production, a high cutting speed and little expenses for support technology.

The RPV has a wall thickness of 124 mm (carbon steel) and a 7 mm inner cladding of stainless steel. Its total height is 14.5 m and the inner diameter is 3.7 m. Manufacturing tolerances of 20 mm have to be compensated by the dismantling device. Moreover, consoles for RPV internals (e.g. for grid plates, core shroud, etc.) have to be considered in the planning. **Figure 5** shows the actual decommissioning status of the RPV at KRB A.

Applying cutting techniques in air may result in a strong generation of radioactive aerosols, which has to be managed by means of additional health protection equipment. A water overhead avoids spreading of the radioactive aerosols. Moreover the water overhead shields the working staff from high dose rates. Therefore, mechanical cutting under water provides the best radiological protection. But, of course, it is not possible to cut through the RPV wall without lowering the water level.

The possible mechanical procedures for dismantling the RPV are circular sawing, band sawing and milling. The implementation of mechanical cutting techniques requires strong tool carriers that resist the restoring forces and vibrations. The tool support has to be fastened tightly to the RPV so that a 360°cut can be managed without any disarrangement of the machine. Mechanical cutting of 131 mm steel entails high tool wear, except when using a band saw. Because of the geometric boundary conditions of the RPV dismantling, a band saw can only be used for post-dismantling purposes. The advantages of band sawing are good cutting performance and a high tool life.

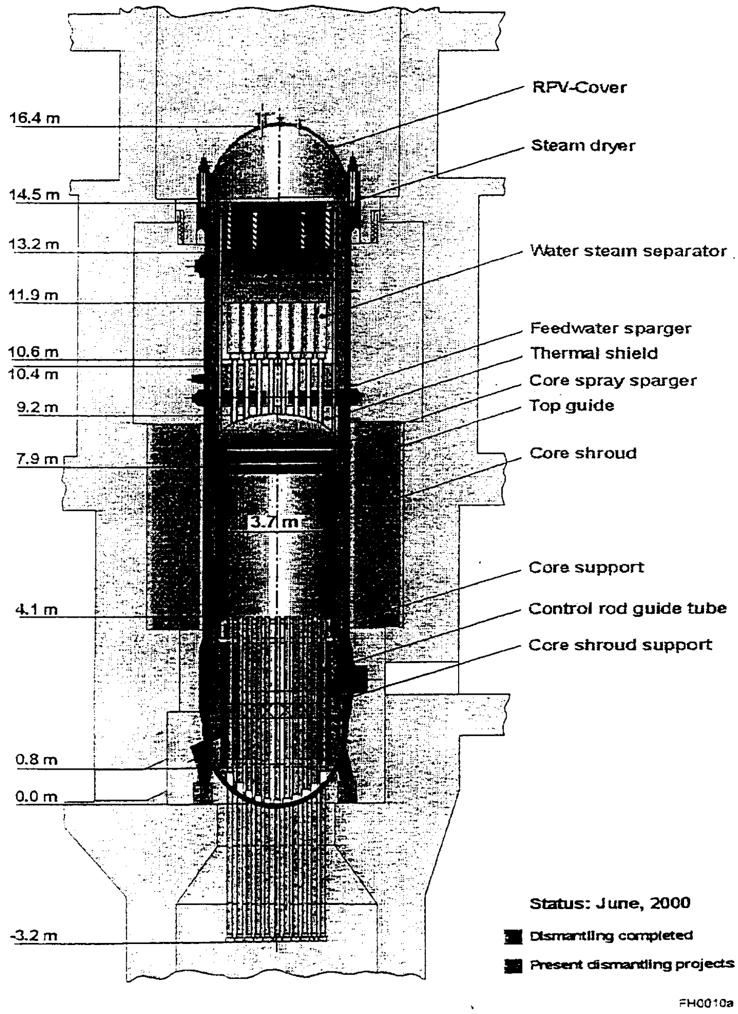


Figure 5: Status of dismantling the RPV in KRB A

Because of strong driving and restoring forces the circular sawing technology is problematic. A breaking of the tool causes serious problems when the sawing blade is wedged in the RPV. The whole machine has to be lifted and the cut must start at a new position.

The advantage of milling is the ability to cut thick materials with only little risk of the tool wedging in the kerf. The milling tool is a strong tool with a relatively high tool life.

Thermal cutting has the advantage of high cutting speed and no restoring forces. Therefore the tool carrier can be of a more simpler design.

Oxy-acetylene flame cutting of the vessel is only possible when cutting from the outside to the inside, because the austenitic cladding can only be melted from the ferritic side. With plasma and CAMC-procedures it is possible to cut through the RPV wall from both sides. The disadvantages are strong aerosol generation when using plasma cutting, and lots of slag when using CAMC. Both procedures can be used to cut a gap in the RPV from the inside, especially to cut through the austenitic cladding. The highest cutting speed can be performed with the oxy-acetylene torch.

Dismantling studies

Two companies, the consortium of DETEC-MAN and the NOELL company, have been commissioned to develop a dismantling and handling study for the RPV.

The basic idea of both studies is to cut the vessel into ring segments and to transport these segments to the storage pool or a special cutting cabin for post-dismantling into smaller pieces for packaging.

While the NOELL concept is based exclusively on mechanical cutting technique, the DETEC-MAN concept applies a combination of mechanical and thermal dismantling tools. **Figures 6 and 7** give an overview on the applied techniques and the cutting positions for dismantling and post-dismantling for each concept.

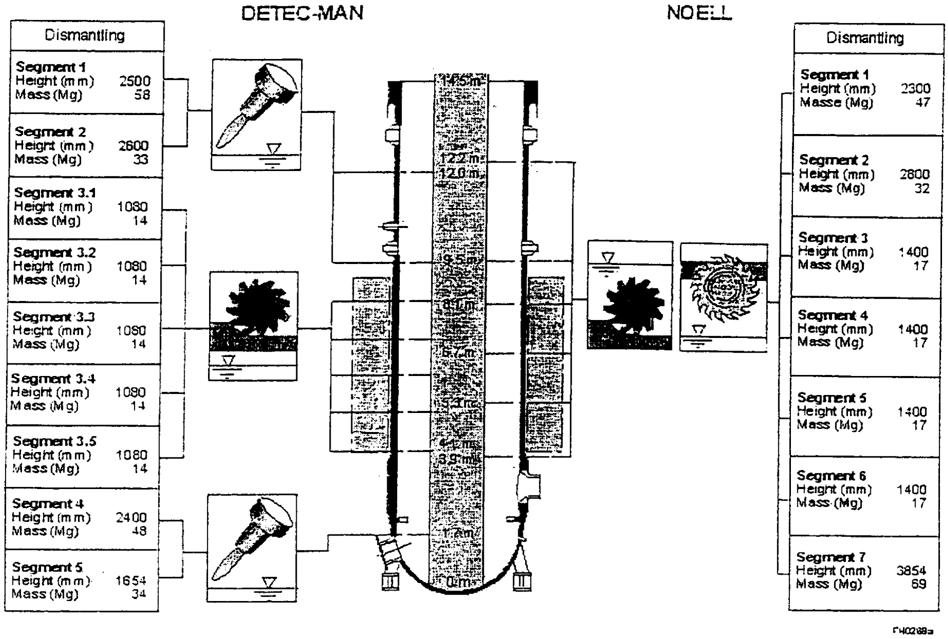


Figure 6: Comparison of two theoretical dismantling concepts

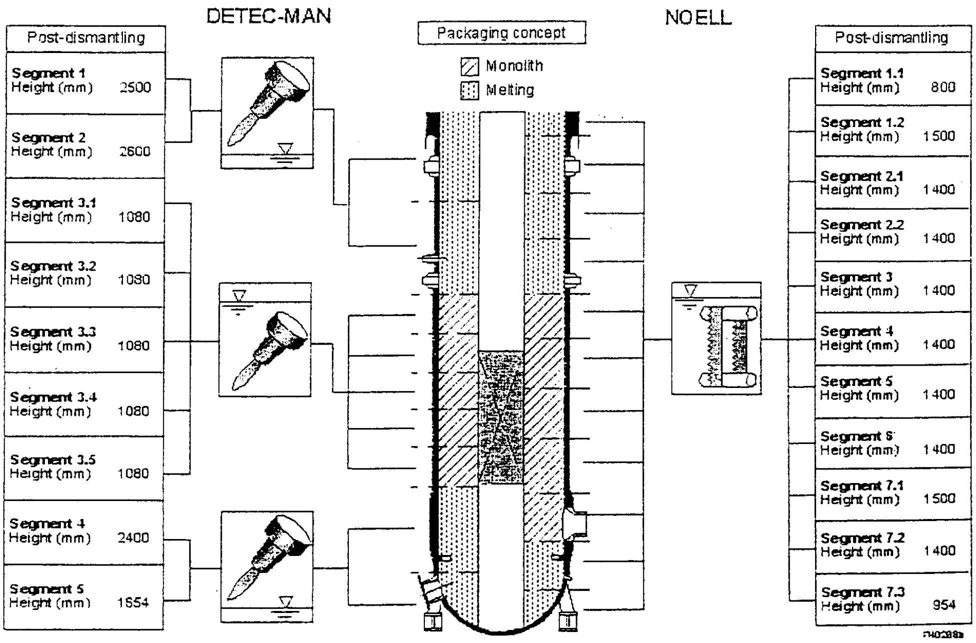


Figure 7: Comparison of two theoretical concepts for post-dismantling

Thermal as well as mechanical cutting can be used for the upper and lower part of the vessel, because the activity is rather low, but segmenting in air with thermal cutting tools is only possible with an effective suction and filtering device.

Remote-controlled mechanical cutting is foreseen for the higher activated core section in both studies. The strategy is to use the water for shielding purposes as long as possible. The disadvantage of using mechanical cutting tools is that these tools ask for heavy tool support, which must counteract the reaction forces. Heavy tool support at the same time will involve in high expense.
The main topics and differences of the two concepts are explained below.

The DETEC-MAN concept

This concept plans to cut the RPV in 9 ring segments by combining mechanical and thermal cutting.

Thermal cutting in atmosphere is selected for the low activated upper and lower RPV areas. Manual positioning of the thermal cutting devices in this case is acceptable. For the highly activated RPV wall in the core area, a remote-controlled mechanical technique is proposed. Because of the wall thickness of 131 mm, the chosen technique is milling.

Prior cutting the RPV, the walls will be cleaned intensively. It is foreseen to do the cleaning within the RPV by high pressure water jet. The core shroud manipulator will guide the nozzle.

An acetylene torch which is fixed on a chain driven tractor, running around the RPV on a rail will execute the thermal cutting. To reduce the aerosol emission the RPV is always filled with the maximum possible water level during cutting. The torch has an inclination of about 30° so the cutting flame entry is above the water level and the exit is under water. By this procedure the cutting depth is increased from 124 mm to 151 mm, but the slag is blown into the RPV water. This reduces the spreading of aerosols and will have a positive effect on the situation of radiological protection for the staff.

During thermal cutting a suction cover will be installed at the top of the RPV, so that all emitted aerosols can be collected. During the cutting progress, wedges will be manually pushed into the gap between the RPV and the cut off segment in order to keep the segment in position. A traverse lifts the segments to the post dismantling area in a temporary tent on the reactor floor.

A milling unit is foreseen for the high activated core area of the RPV, (**Figure 8**). The machine is positioned inside the RPV and fixed by clamping devices at the wall. Then the cut is done in atmosphere. During milling, wedges have to be inserted into the cut to fix the segment in its position. The cutting process is visually and acoustically controlled to avoid a tool damage. A suction hood on top of the RPV is needed to suck-off radioactive aerosols.

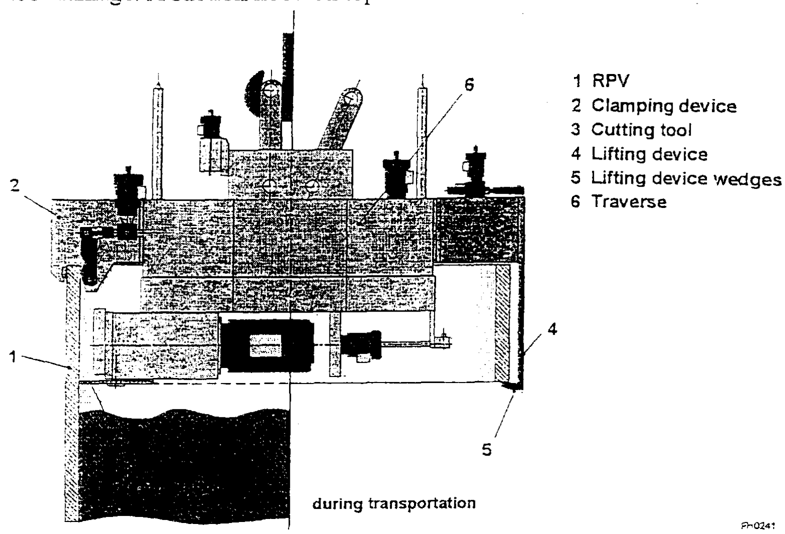


Figure 8: Conceptual study for a segmenting unit (Detec/MAN)

The post-dismantling of the low activated parts will be done at the reactor floor in a special cutting tent. The rings will be dismantled by oxy-propane cutting into small segments which can be packed in 20′-containers. Because of the application of a thermal cutting technique, no device for taking up the restoring forces is necessary. The cuts will be done vertically, a fixing system prevents tipping of the segments.

The higher activated segments will be post-dismantled in the fuel element storage pool. During transportation in air they are shielded by a hood. The segments will be positioned on the turntable which is already at hand. A remote-controlled thermal cutting unit will be used for dismantling the ring segments into 9 single plates for subsequent packing into onion cast containers. To allow safe transportation to the underwater packing podest, preliminary cutting of holes is necessary to attach an arresting device.

Before the bottom dome will be removed from its installation place the penetration pipes of the control rod drives have to be cut off with a dressing tool that is positioned inside the pipes. The cut off pipes are caught by a basket that is positioned directly below the bottom dome.

The DETEC-MAN concept investigated the packaging by conventional cast-steel containers and the new onion casting technique. All together 249 Mg of RPV-material has to be handled. 6 Mg of swarf material will be produced.

The upper two and the lower two parts are considered as meltable because of their low activation. The parts of the core section (level 12.9 m - 18.3 m) are highly activated and are foreseen for final storage. It is proposed to put them in onion cast containers and to fill these containers up with material from one of the other parts. The mass for final storage is calculated to be 67 Mg. Additional 48 Mg of RPV-material with low activity will be necessary to fill up the onion waste containers. In total, 15 onion cast containers will be needed.

The conventional packaging variant uses for the low activation normal containers with a concrete fixing and for high activation the steel-cast containers. The volume for final storage by this variant is 64.5 m^3. The onion casting variant needs 15 cast containers with half the size of a conventional Konrad container. The parts of segment 3 will be packed into the containers and fixed with the molten segment 4. The volume for final storage by this variant is 40.8 m^3.

The NOELL concept

In the NOELL concept a combined milling and circular sawing technique is proposed for segmenting the RPV into seven parts. A pre-cutting of the RPV by horizontally sawing under water is followed by reducing the water level for completing with the final cut in atmosphere.

First a gap is cut into the RPV wall with the milling tool up to a remaining wall thickness of 11 mm. It is foreseen to execute this in two steps by the use of milling tools with 60 and 40 mm diameter. After this, the water level is lowered and the final cut is done with a circular saw. The gap created by the milling steps is wide enough to allow sawing.

The advantage of milling is the ability to cut thick materials with only little risk of the tool wedging in the kerf. The residual wall thickness of 11 mm can be cut without high restoring forces and with minimal danger of a tool damage by the circular saw.

The tool life of the milling cutter is estimated to be 20 min for stainless steel cladding and 60 min for the ferritic base material of the RPV. Due to these given values, a lot of tool changes will be necessary. A disadvantage is, for a tool change the total cutting equipment has to be lifted to the reactor flange level. To minimise the consuming costs reversible cutting tools are used.

The dismantling unit is divided into two parts: the lower ring type tool carrier and the upper fixing and handling traverse. The tool carrier is equipped with a turning plate, a drilling unit and a milling unit. The traverse is equipped with a fixing device to handle the segment safely. **Figure 9** shows the dismantling unit.

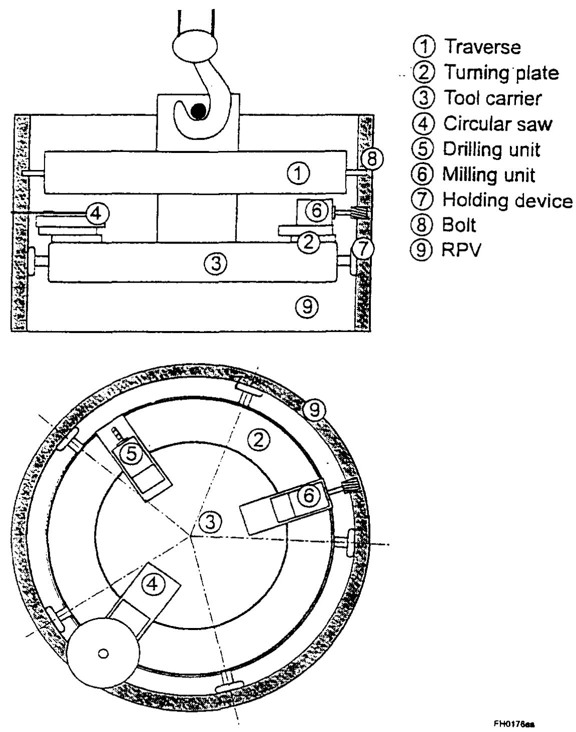


Figure 9: Conceptual study for segmenting unit (Noell)

The unit will be positioned in the RPV for cutting. Holding clamps (7) will keep the device in balance, so that the drilling unit (5) can drill three stop holes (8) for locking the traverse (1). When the traverse is locked the milling unit (6) will cut about 80% of the RPV wall under water. Afterwards the water level will be lowered and the circular saw (4) will cut the residual 20% of the RPV wall in atmosphere.

The segment will be transported after cutting with the traverse to an underwater turntable in the fuel element storage pool for post segmenting with a band saw. During transportation the tool carrier will stay in the RPV. The segments 1, 2 and 7 of the RPV will be post-dismantled to reach an appropriated size for packaging. Segments 3 to 7.1 of the core area are foreseen for final storage.

The post-dismantling for all segments is performed in the fuel storage pool by a special under-water band saw with good cutting performance for thick walls. By a simple manipulation the band saw is able to perform vertical and horizontal cuts without changing the work pieces position.

For handling the post segmented RPV parts, boreholes e.g. for the attaching arresting devices will be cut with a CAMC-(Contact-Arc-Metal-Cutting)-machine.

The penetration pipes of the control rod drives are cut off with a hacksaw. To reach the pipes, a scaffold is built-up in the control rod drives room. The nozzles at the RPV outside are cut off with a dressing tool.

Conclusions drawn from both studies

From the technical point of view the use of mechanical tools would ask for heavy support to counteract the reaction forces during cutting. Therefore the expense for this technique is high. Other disadvantages are the complexity of the support and driving systems and the short tool life. On the other hand mechanical tools are well known standard tools, producing nearly no aerosols but filings, being easy to collect.

The main advantage of thermal cutting techniques in opposite is the flexibility of the tool. Cutting can be performed with high speed and simple tool carriers also for non-linear tracks. Therefore a standard oxy-acetylene torch is without any competition from the economical point of view. But it has to be taken into account that aerosols from the molten and evaporated material will be spread in the atmosphere. Experience from other projects, e.g. thermal cutting of the RPV-head, has shown that this problem can be controlled.

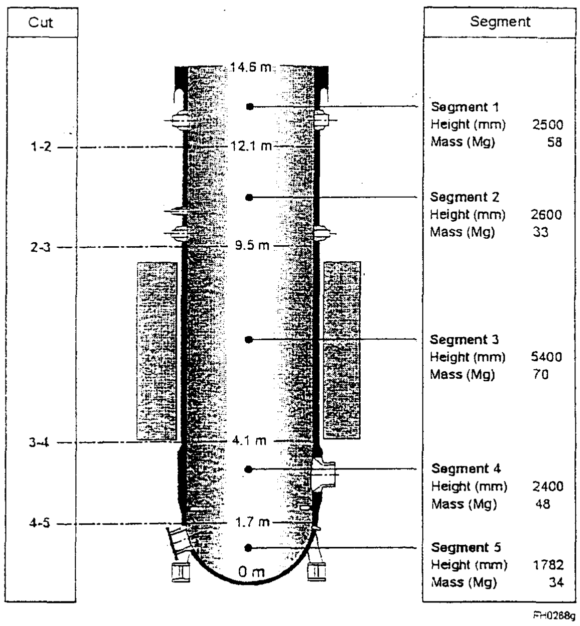


Figure 10: Foreseen dismantling strategy by thermal cutting tools

Actually it is foreseen to dismantle the RPV exclusively by thermal cutting tools. Cutting will be done by oxy-acetylene flame cutting from the outside of the vessel to the inside, because the austenitic cladding can only be melted from the ferritic side with this technique.
The RPV will first be dismantled into five ring segments (**Figure 10**) by an acetylene torch, which is guided around the vessel on a rail. The torch has an inclination of about 30° so that the cutting flame enters the material in a position above the water level whereas the slag will be blown out of the kerf under the water level (**Figure 11**). This will reduce the amount of aerosols.

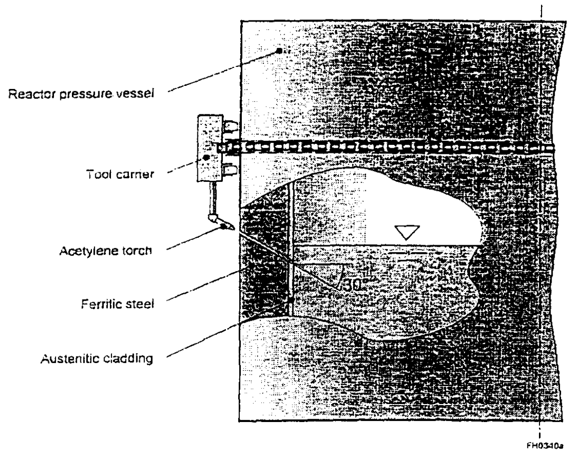


Figure 11: Cutting the RPV by an inclined oxy-acetylene torch

Thermal cutting in situ is only acceptable for the low activated upper and lower RPV areas. Due to this, the high-activated core area of the vessel will be removed in one piece without a cut in the core zone. This core segment is 5.4 m high, its weight is about 70 tons and it can still be handled with the reactor building crane.
Post-dismantling of this RPV segment into five further ring segments follows by thermal cutting under water in the fuel element storage pool. To comply with the licensed weight for the pool, the water level can be lowered if necessary. After vertical segmentation of the rings, the dismantled material from the core zone will be packed into onion cast container.

The other segments of the low activated area will be post-dismantled by thermal cutting in a closed cabin.

B.2.2. Support positioning and driving system

<u>BR3</u>

The desolidarization and the lifting of the RPV give us the possibility to dismantle it in the same way as we did for the internals: using the refuelling pool as underwater workshop.

Figure 12 shows the refuelling pool during RPV dismantling. The main support, positioning and driving systems are:

- ***Support for turntable***

It is a heavy steel construction (design load of 40 t) that is installed on the reactor vessel support ring after removal of the vessel. This support must receive and hold the turntable that is the central dismantling component.

- ***Turntable***

The turntable is made of stainless steel. It allows to clamp the pieces to cut and to present them in front of the cutting equipment. It can be driven manually using long handling tools to drive the gearbox or automatically using hydraulic power on the gearbox. On the turntable extensions, cutting equipment and/or additional clamping devices can be installed.

- ***Linear gantry crane***

The linear gantry crane of 40 tons was installed above the pool only to lift the RPV out of its cavity, to position it on the turntable and to remove it for service. During the horizontal cuts, the linear gantry crane always supports the weight of the piece above the cut.

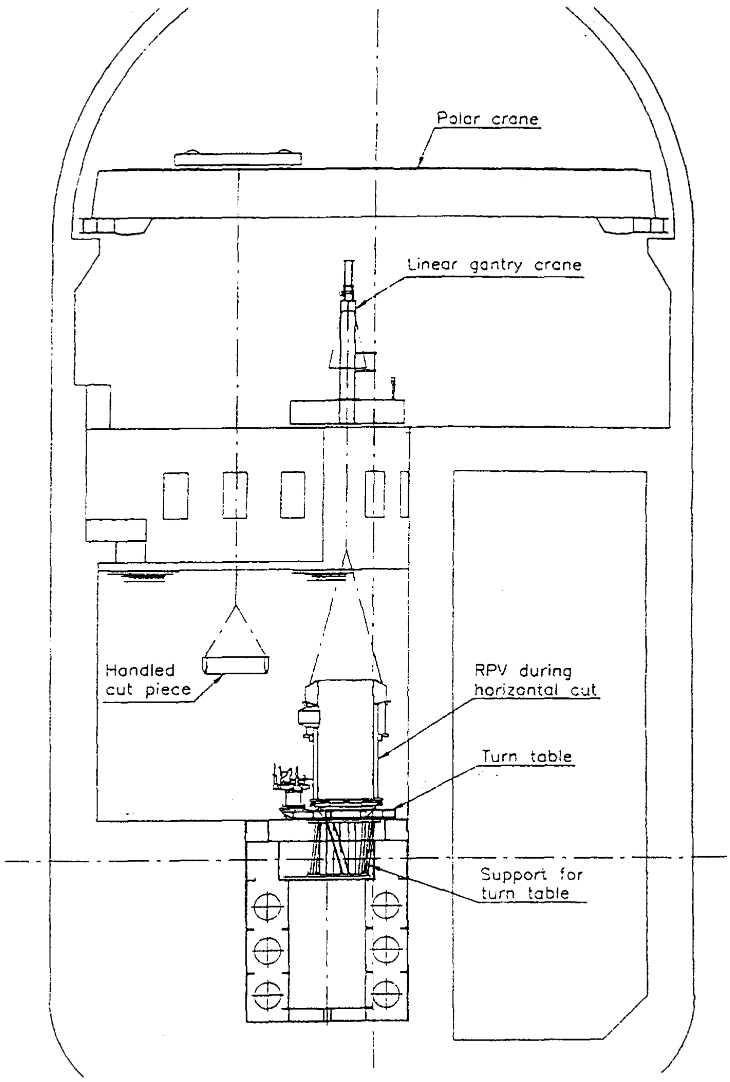


Figure 12: Refuelling pool during RPV dismantling

- ***Polar crane***

This crane was used during exploitation of the plant (1 hook 40 t, 1 hook 10 t). For dismantling purposes it had a new function: tools installation/removal and all other handling operation such as transfer of cut pieces. It is possible to use the linear gantry crane simultaneously with the polar crane.

- *Handling of cut pieces*

The cut pieces are rings and, finally, segments. Rings will be handled using a set of three automatic clamping devices hanging at the polar crane. These tools are adapted from the industry to be activated remotely. For the manipulation of segments, a specific tool was designed in order to move and install them on the storage racks.
For the manipulation of the racks, see B.2.3.

EWN

For the model dismantling of the non activated reactor components of units 7 and 8 in unit 5, several transports with the existing crane facilities and corresponding load lifting devices have to be performed in the reactor hall.

For the transports, the following available crane facilities and load lifting devices will be used:
- bridge crane 250 Mg (reactor hall crane);
- bridge crane 32 Mg (reactor hall crane);
- grab tool with reactor protection container 71/35 Mg.

The following load lifting devices which have to be newly manufactured are foreseen:
- transport traverse for the RPV, lifting capacity 250 Mg including five screwed bolts M 140 x 6 which connect the shielding and transport traverse and the RPV;
- load lifting devices and load supporting points of the assembly "Load induction SG-mounting hatch of the wet cutting caisson";
- coupling piece for the transport traverse RPV with connecting bolt (lifting capacity 250 Mg);
- traverse for the horizontal transport of assemblies, lifting capacity 250 Mg;
- lifting and lowering device dry cutting caisson;
- transport crane wet cutting caisson (16 Mg);
- traverse for container.

The reactor protection container (**see figure 13**) with grab tool has the following functions:
- transport of the protecting tube system, core basket and reactor cavity with cavity bottom;
- biological protection of the operational personnel.

The protection container consists of 5 thick-wall cylindrical strakes. The strakes are connected with each other by bolts. The container is handled with the existing 250 Mg overhead crane.

By the dismantling of the core basket and the protecting tube system, the protection container is set down on the reactor flange. The grab tool goes by crane through the protection container to the part to be transported. After the interlocking, the core basket or protecting tube system is drawn in the protection container and transported to the cutting place.

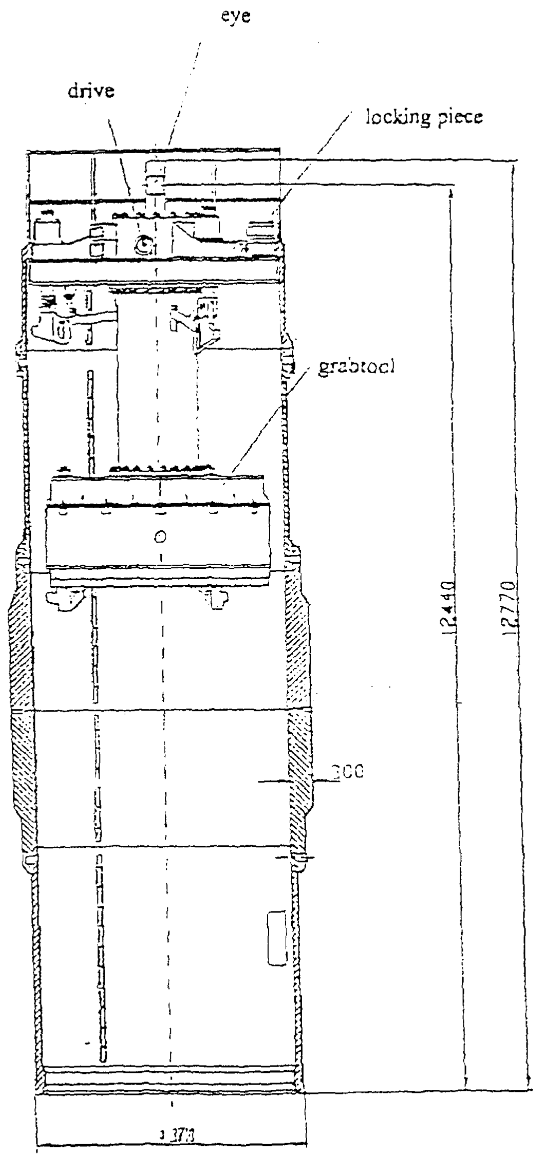


Figure 13: Reactor protection container

KRB-A

System for dismantling the core shroud

For dismantling the core shroud, a special machine was developed (**Figures 14–15**). This self-centring manipulator has four arms and is designed for being positioned on top of the core shroud as well as being centred within the RPV. The machine can easily be adjusted to various diameters by changing the length of the arms. The central manipulator arm works in cylindrical co-ordinates. It is possible to fix a plasma torch for cutting or to mount a brush or high pressure water jet for cleaning tasks to this rotating arm. For cutting tasks, the machine has been modified to perform radial and feeding movements simultaneously. This is necessary to maintain a constant distance between the torch and the wall during the cutting process.

Moreover, pneumatic cylinders were mounted at the arms of the machine, to control the integrated load hooks remotely from the reactor floor. By this, it is possible to lock the cylindrical segments after cutting and to unlock after positioning on the rotating table in the post-dismantling area. The hooks are connected to the crane with the main bearer cables, so that the dismantling unit does not carry the load itself. The design could therefore be of a slight type.

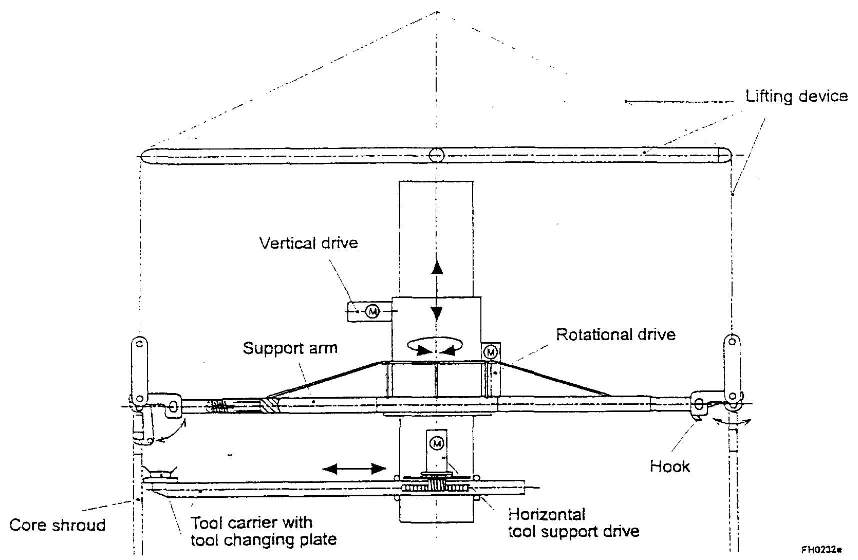


Figure 14: Sketch of the dismantling unit for core shroud

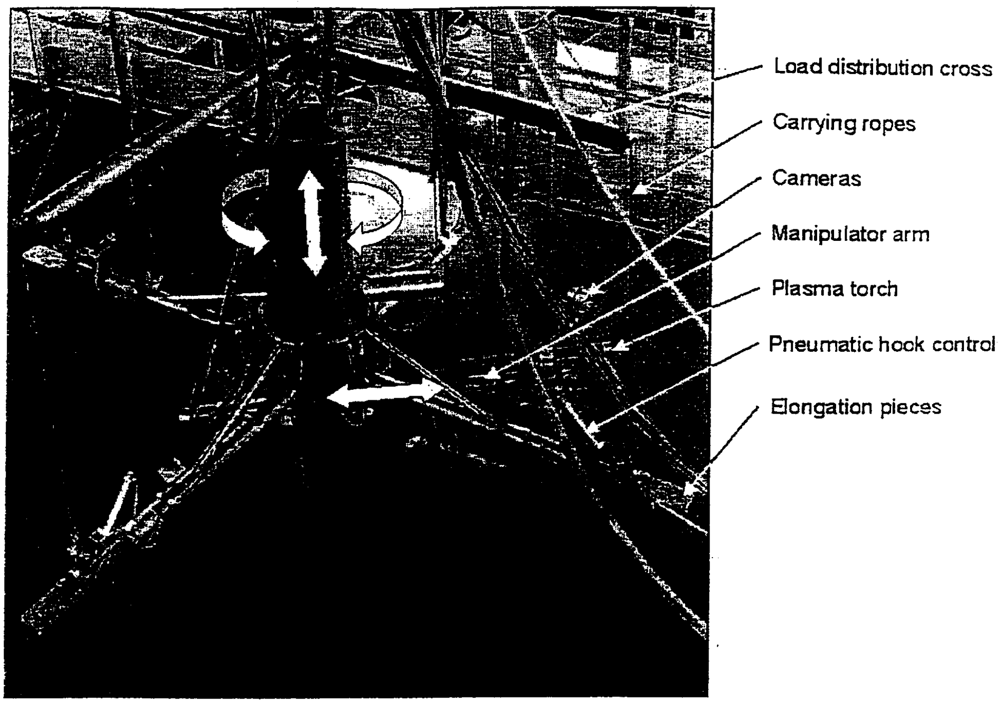


Figure 15: Core shroud dismantling unit above the reactor pool

After positioning the system on or in the core shroud the central arm drives the plasma torch beneath the four arms for cutting small windows in the wall of the core shroud. Then the integrated hooks can be locked. Afterwards the central arm drives the torch to the cutting position, which is given by the segment length, e.g. 800 mm. By rotating the central arm with the plasma torch about 360 °, the segment is cut off and can be lifted to the post-dismantling area.

The control unit (**figure 16**) was modified, in view of having all operating elements in a favourable position to the supervisor. This includes the operating elements for the cutting device as well as the control unit for the plasma torch and the monitors of the underwater cameras.

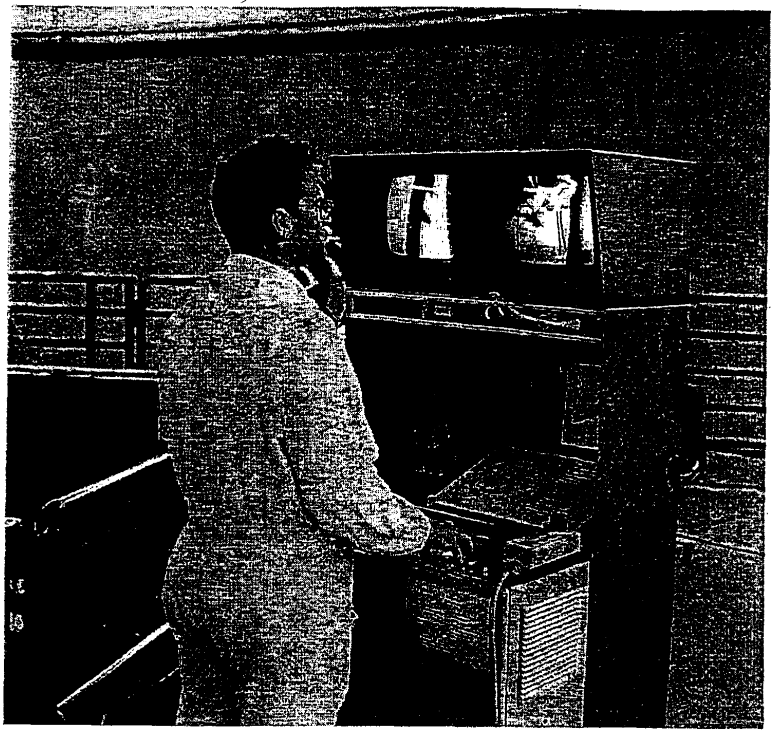


Figure 16: Control unit for the cutting device

Systems for dismantling and post-dismantling the RPV

Heavy supports will be necessary for mechanical cutting or for moving the ring segments to the storage pool or a special cabin for post dismantling. The existing crane in the reactor building can be used for driving and positioning these supports. In case of separating the segments from the cutting device, an additional transport system must be used, or the cutting device must be able to fix itself to the pressure vessel.

The DETEC-MAN concept for dismantling the RPV proposes to use an oxygen torch tractor, which moves orbital on a magnet circle system at the outer wall of the RPV. The magnet circle system has to be fastened manually for every cut. The torch is inclined at 30°, so that the flame is directed downwards. The water level is held a little lower than the kerf.

The higher activated parts will be dismantled mechanically to avoid strong aerosol production. For this purpose a special traverse with a tool carrier has to be manufactured (**Figure 8**). The segmenting unit is fixed at the RPV by 4 clamping devices. A milling device that is adapted on a turning arm does the cut. The milling unit is positioned below the tool carrier. This makes it possible to combine the tool carrier and the segment carrier in one unit.

The disadvantage is a bad reachability of the milling tool when it has to be changed. It is necessary to prop the cut off segment by wedges to protect the milling tool.

For post-dismantling, the DETEC-MAN association proposes thermal cutting of the ring-segments in the fuel element storage pool in combination with the turning table, which is already at hand at KRB A.

The dismantling unit of the NOELL concept for dismantling the RPV consists out of a tool carrier with three tool holders and a traverse which is positioned upon the tool carrier (**Figure 9**). 5 clamping cylinders centre the tool carrier. Additionally it is fixed by three hydraulic bolts because of the strong restoring forces during the dismantling process.

The traverse is also fixed by 3 hydraulic bolts. With these bolts, the ring segments are transported.

There are tools for milling, sawing and boring on the tool carrier. Boring has to be performed to get holes for the fixing bolts. During boring the holes the machine is fixed only by the clamping cylinders and the crane.

The tool devices are positioned between the upper bolts of the traverse and the holding devices of the tool carrier. By this arrangement blocking of the sawing blade during cutting can be avoided.

For post-dismantling, in the NOELL concept a band saw is placed in the fuel element storage pool. The segments are fixed on a turning table, which is automatically driven and controlled by an angle-measuring instrument. A special turnover device is used to bring the segments in horizontal position for consequent packaging into cast containers.

A special tipping device is used to segment the bottom dome. It is placed upon the turning table and enables to perform all necessary cuts at the bottom dome. The dome is fixed with a jack and two bolts on the tipping device. An axe with two bearings allows to turn the bottom dome to a vertical position. The band saw is able to perform horizontal and vertical cuts without getting into conflicts with the control rod penetration pipes.

B.2.3. System for the collection and filtration of swarfs and debris

<u>BR3</u>

During the dismantling of the RPV, one had to collect two totally different types of (secondary) waste. On one side, one had to collect the metal swarfs that were produced with the cutting techniques and, on the other side, one had to collect the thermal insulation that surrounded the RPV. A new bought filtration system collected the swarfs while, for collecting the insulation, a lot of different techniques were applied.

A. The cyclone filtering system

Due to some shortcomings (e.g. multiple swarf blocking, a low filling factor of the waste transport container,...) the increasing wearing (e.g. the sealings) and the high contamination grade of the old filter installation (after 6 years staying under water), the dismantling team decided to buy a new filter installation with better adapted properties.

The new water filtering installation works following the cyclone principle. Here, a conical piece (the cyclone) forces the water into a downward spiralling movement. Due to the radial forces and the gravity, the swarfs ($\rho > 1$ kg/dm^3) are separated and moving downwards. The main advantages are the lack of pressure drop on the suction side of the pump and the easy collecting of the swarfs at the bottom of the cyclone. The lighter swarfs ($\rho < 1$ kg/dm^3) are trapped in a fine filter situated after the pump.

Figure 17 shows the main subassemblies of the filter installation. The heart of the installation is the pump with its pump housing (1). Underneath is situated the cyclone (2) where the separation heavy/light swarfs takes place. A collecting basket (3) under the cyclone gathers the separated swarfs. After the pump, there is a special flange for the connection of a filter bag (4) or a filter housing (5) with five 20" coarse filters. The whole installation rests on a support (6). A 2"1/2 hose connects a suction device (7) to the filtration installation. This device is provided with one of the three suction mouths (8a, 8b and 8c).
The whole installation is built out of stainless steel and can be used out of or under water. This implies that all the main parts can be removed and handled remotely under water with a special long handling tool (9).

The advantages of the new filter installation are:
- The whole installation can be disassembled remotely under water with a long handling tool.
- The maintenance can be carried out quicker and easier. The part that causes a problem can be taken separately out of the water.
- The trapped swarfs do not cause a pressure drop and are collected in a removable collecting basket.

- The collecting basket can easily be taken away and the swarfs can be dropped in a greater basket that increases the filling factor of the transport container.
- The same applies for the fine filters. Because it is possible to take the coarse filters one by one (and not as a whole filter body), they can be put in a special transport rack to increase the filling factor of the transport container.
- The purchase cost of the filter elements reduces: one great collection basket against several strainers and single coarse filters against filter bodies, with repartition plates.

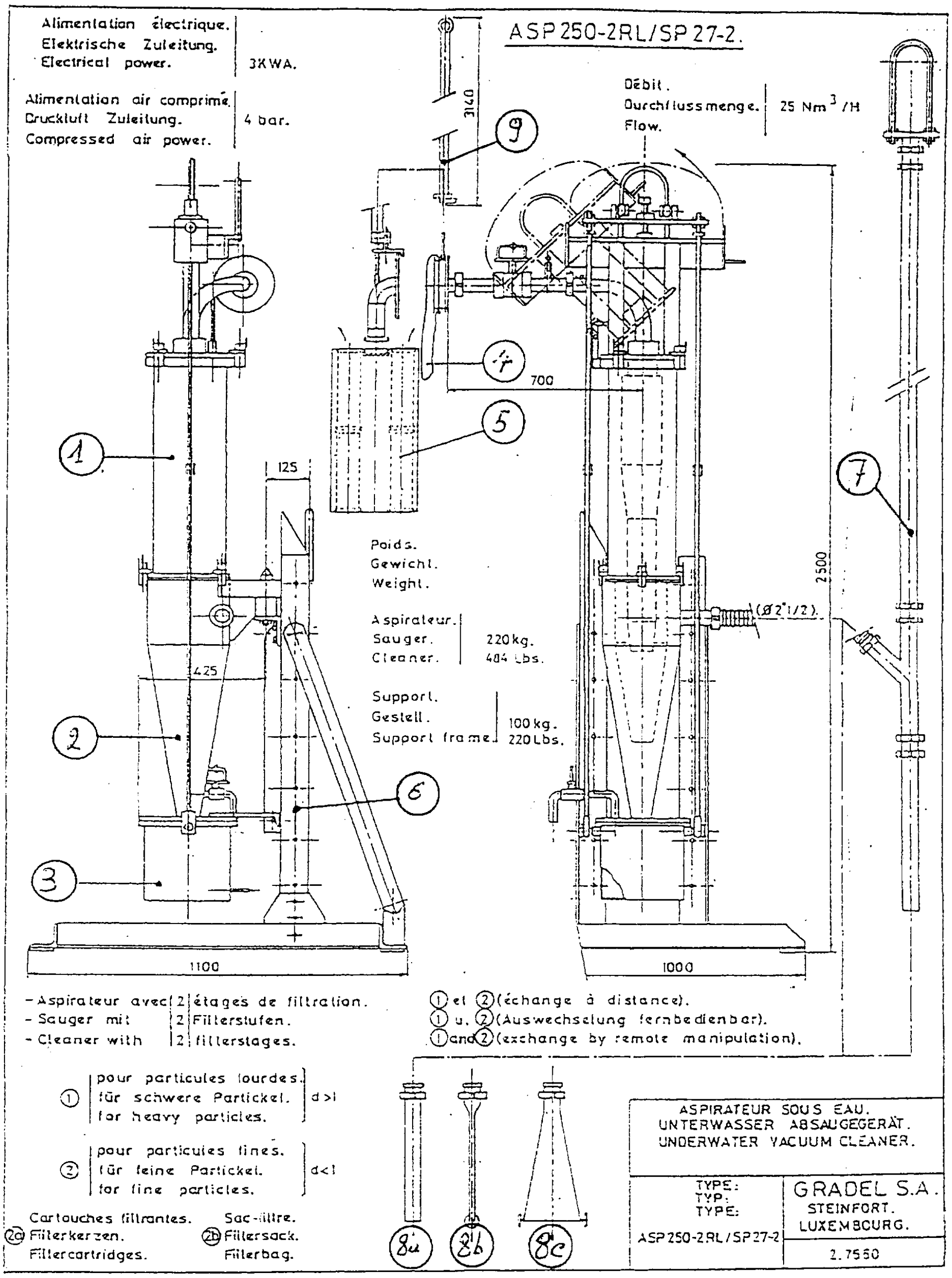


Figure 17: Main subassemblies of the filter installation

Although the cold tests at the factory were promising, a lot of swarfs plugs were created in the filter system during the horizontal cutting. They were mostly situated in the conical part of the filter housing. A possible explanation is that the milling process produces long and curled swarfs. In the filtration system, these swarfs have to pass a narrow gap between the conical part of the housing and the protective strainer of the pump. Due to the specific form of the swarfs, they could easily cling together, forming a plug in the narrow gap. To reduce the number of these plugs, one installed a collecting drum on the suction side of the pump (see **figure 18**). This drum is foreseen with a trapdoor at the bottom. This is used for immediate disposal of the swarfs in the waste drums. The trapdoor can be remotely controlled in case of the underwater unloading of high-activated swarfs.

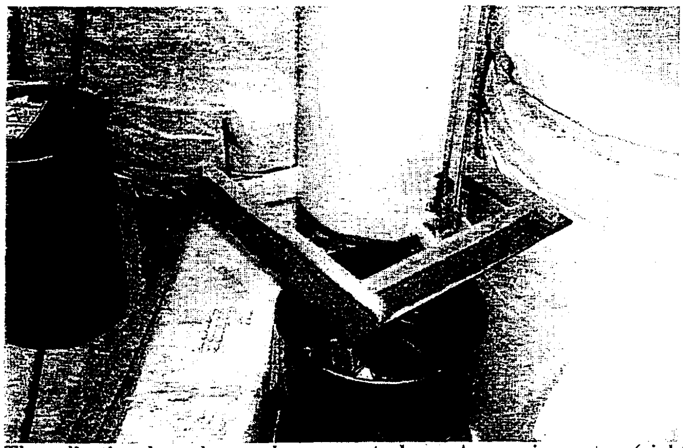


Figure 18: The collecting drum hangs above a waste drum. An opening system (right from the collecting drum) opens the trapdoor at the bottom to release the swarfs that were collected in the drum.

B. Collecting of the thermal insulation

For the collection of the thermal insulation, not so highly radioactive and heavier than water, one used different techniques.

- *The main filtration system of the pool*

The heart of this system is a filter containing 64 10"-cartridge filters of 1 μ and two demineralizers. The filter capacity goes up to 30 m^3/hr. With this system, the small insulation particles could be trapped. The effectiveness of this system was proven during the encountered visibility problem caused by this insulation (more about this visibility problem in chapter B.4.)

- *External pool filtration system*

To increase the filter capacity of the pool during the visibility problem (see chapter B.4), one installed an external filtration system. This system has a capacity of 20 m^3/hr and has 10 30"-

cartridge filters of 1 μ and at the end there are two demineralizers. One has the capacity of 200 l, the other 150 l.
The used filters had a fine layer of fibres at their surface, which means that also this system contributes to the collecting of the thermal insulation.

- *Collecting net*

Before the installation of the RPV on the turntable, one installed a safety net between the RPV and the turntable. Along the protection of the moving parts of the table, the net could also collect the sunken insulation. By pulling the net out of the water, a great part of the sunken insulation could be immediately removed out of the water.

- *The pneumatic gripper and plunger pump*

First, a pneumatic gripper took the remaining big parts of the insulation. This technique was not successful. Due to the low density of the fibres they drove away with the water movements caused by the functioning gripper.
Instead, the operators used a simple plunger pump to collect the largest parts of the sunken insulation. Using the underpressure at the suction side of the pump, they could grip the biggest insulation parts and put the insulation in a collecting basket. Afterwards, the basket came out of the water and the insulation could be dried and disposed of as supercompactable waste.

- *The pool cleaning robot*

Another selected technique for collecting the sunken insulation was the use of a common swimming pool-cleaning robot. Cold tests showed out that the robot was capable to collect thermal insulation. The robot is foreseen of a 40μ-filter bag and can function in a teleoperated mode or in an automatic mode. The robot is mainly used to clean the swimming pool bottoms but it can also climb against the pool walls for cleaning them.
The robot has done a good job, but nevertheless it has two main disadvantages:

- The finest filter bag is only 40μ. This means that only large particles such as long insulation fibres can be collected but not fine particles. On the contrary, the presence of those small particles causes a cloud of 'dust' behind the robot that strongly reduces the visibility of the pool water after using the robot for a longer period.
- Because it is a common cleaning robot, one could not change the filter remotely and/or under water. Therefore, one had to be very careful not to collect high or medium activated swarfs.

Figure 19 shows the different filtration systems available at BR3.

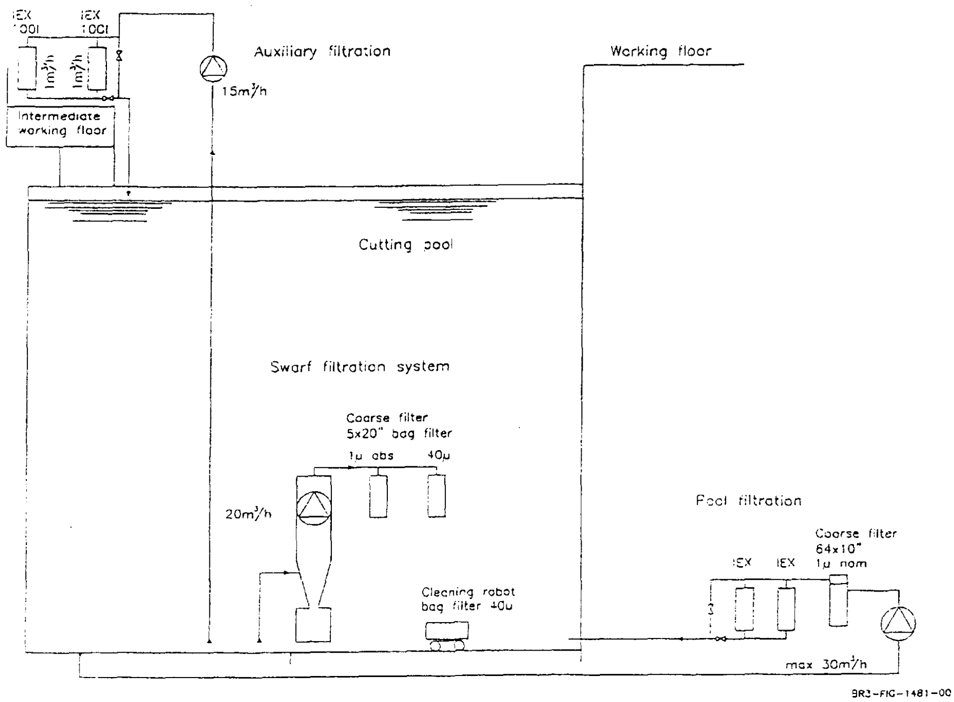


Figure 19: Different filtration systems available at BR3

EWN

Additional ventilation systems

The task of the systems is to suck filter dust, metal fume, carbon black aerosols and fuel gas produced in the frame of the remote dismantling of the reactor components in the steam generator room and to transport the cleaned exhaust air to the central exhaust air system for controlled release to the environment.

This exhaust air system generated a pressure difference between the cutting places and the accessible areas. The accessible areas are directly supplied from the external air system.

The additional exhaust air system generates a directed air stream from the accessible part to the cutting places and also a direct exhaust from the cutting places. The direct exhaust air from the cutting places is led through a cyclone separator. The cyclone separator separates the heavy particles from the air. The exhaust air then passes a HEPA filter, where 99.997% of the airborne particles are separated. There are three filters from which two are always operated and the third is in standby position or cleaned.

From the accessible areas (rooms), the cleaning of the exhaust air streams works according to the same principle, whereby a cyclone separator pre-cleaning is not necessary.

Water cleaning system

The cutting pond has connections for water feed supply and discharge. The designed minimum filling value for the water cleaning system is ca. 270 m^3, the maximum value 390 m^3. During the cutting work, the water can be circulated and cleaned with the water cleaning system in the pond. The water cleaning system is necessary to maintain the required water quality in the cutting pond and to reduce the activity concentration of the pond water. The floating particles (material from the partition lines and exhausted material from the cutting tool) and coarse particles depositing on the ground will be released from the pond and separated in a filter system.

For the design of the water cleaning system for the wet cutting caisson, the following reference values were established (see **table 3**):

Table 3: Reference values for the water cleaning system design

Cuttings ca.	1.6 kg/h
Portion	99%
Cuttings, design value	2.0 kg/h
Activity	1.1×10^{9} Bq/g[1]
Airborne particles	18 g/h
Portion	0.9 %
Water particles	2 g/h
Portion	0.1 %

Note: 1 average specific ^{60}Co activity for the core basket

The average entry of floating and airborne particles from the CAMC procedure is ca. 70 g/h. The following water values have to be adhered to:

- floating particle content <3 mg/l
- particle size < 3 µm.

Due to radiation protection reasons, the whole water cleaning system is arranged inside the water pond and mounted in a steel framework, through which it is possible to insert the complete module into the pond and lift it out. The filter exchange will be done mainly under water.

KRB-A

In principle different cutting techniques can be used for the dismantling of the reactor pressure vessel.
The experience gained from the cutting of the reactor head showed that the segmenting in air with thermal cutting tools is possible with an effective suction and filtering device.
Cutting under water would reduce the emission of aerosols to a minimum, but this of course is not possible, because dismantling must be done in situ and the water level has to be below the cutting position.
The investigation for working with mechanical and thermal cutting techniques in air and under water has shown that the emissions of the tools can be controlled with the existing filtering systems. The capacity of filtering systems is strongly depending on the cutting technique and the cutting environment. Mobile filtering systems are very flexible and they can be renewed easily. Recleanable systems should be used for stationary systems in air and under water.

The DETEC-MAN concept for dismantling the RPV proposes to use oxygen cutting and to incline the torch by 30°, so that the process starts in air, facing the outside of the vessel, but the slag will be blown out of the kerf right below the water level or at last directly on the water surface. This reduces the spreading of aerosols. It has to be investigated by inactive tests how effective this system will work. Moreover a draining system will be welded at the outer RPV wall to collect slag, water or cooling water from the mechanical cutting.

A covering device is installed on the top of the respective segment. In this cover devices are integrated for suction, for high pressure cleaning and for spray fixing. The suction hood is connected to a high efficiency particle (HEPA) filter system. The high pressure-cleaning device is a rotating nozzle and is foreseen for pre-decontamination of the RPV-walls after the water level has been lowered. Spray fixing is used to fix the remaining contamination before the segments are transported to the post-dismantling area.

The reactor shaft below the RPV flange level is completely covered by a large suction hood to prevent spreading of aerosols during thermal cutting and to enable a defined and filtered air flow. A collecting shell will be installed inside the RPV to avoid strong contamination of the bottom dome by sinking sawdust and cutting slag.

The NOELL concept for dismantling the RPV proposes to clean the inner RPV wall surfaces with a brush that is fixed at the boring unit tool carrier. The hydrosols are collected with a suction unit. Falling sawdust and particles are collected by three sawdust baskets positioned directly below each tool and also by a collecting shell inside the RPV bottom dome. The three baskets are emptied by a pump that is positioned on the machine and connected to a filter.

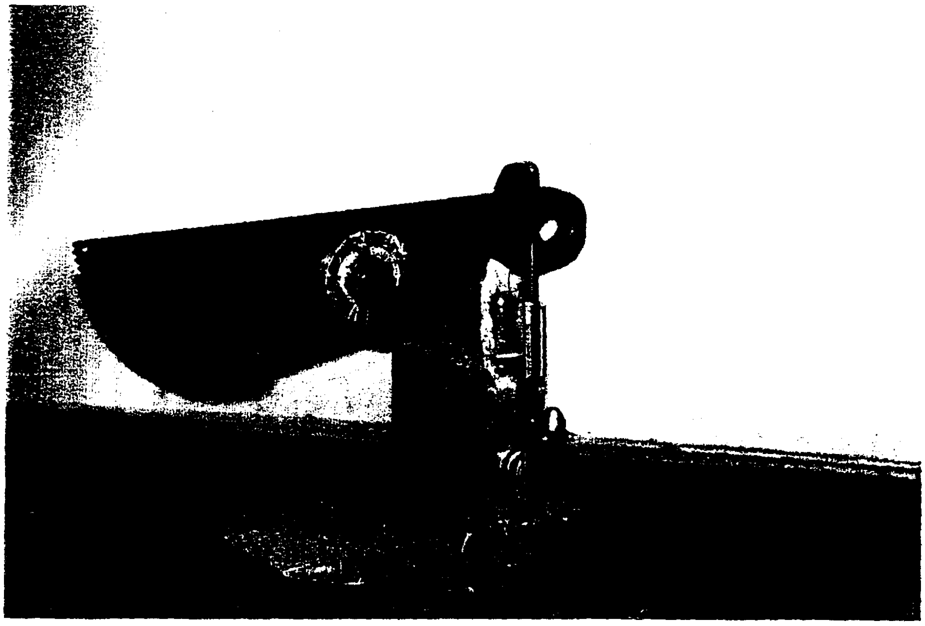
For both post-dismantling principles the same suction device is used. Collecting shells beneath the working place will catch slag, water and contamination fixing means. The low activated rings are decontaminated with phosphoric acid inside the post-dismantling cabin.

Cleaning of reactor water

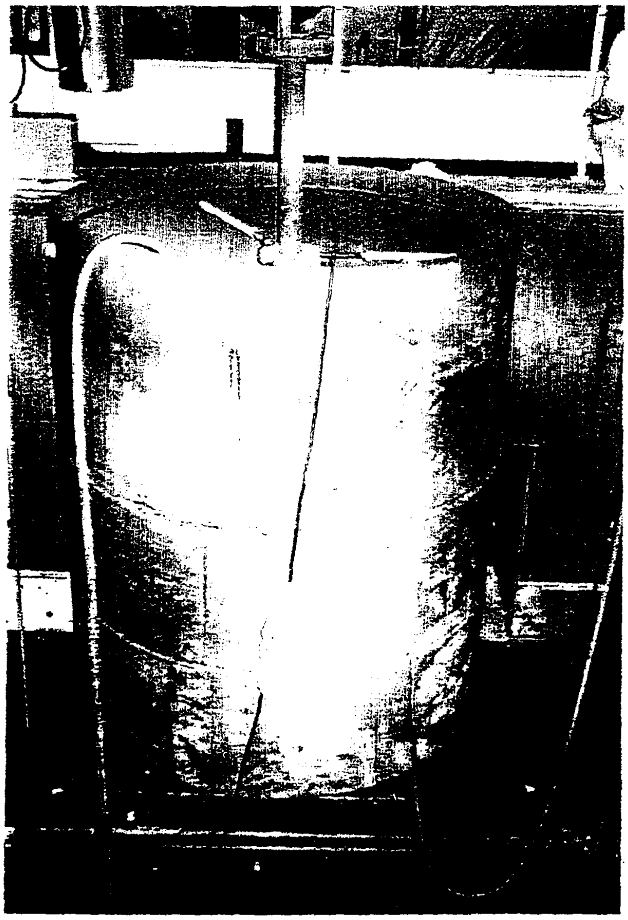
During dismantling of the RPV and its internals radioactive particles from segmenting and mobilised contamination layers will be released into the water.

Bigger parts and slag sink down on the pool floors or the RPV bottom dome and will be collected by a special mobile suction device, whereas suspended particles are removed by the filtering system for the reactor water, which is shown in **figure 20**. The fuel element pool, reactor pool and the steam dryer storage pool are filled with about 1400 m^3 water. Filter 1 is the former operational filter for cleaning the fuel element pool. It is a precoated filter with a maximum flow rate of 30 m^3/h. Resins and celite are used as standard filtering aids.

Although the quality of pollution in the water changed with the starting dismantling of the reactor components, this filter was still sufficient for cleaning the water. However, for improving the circulation within the pools, this filter was supported by a reversible flow filter (Filter 2), which is connected directly to the RPV bottom. The power of this filtering circuit was not sufficient to guarantee constantly clean water, so the arrangement had to be changed. It was decided to replace the filter cartridges and also to modify the piping of the RPV water cleaning circuit partially to achieve a higher water flow rate. A new, stronger pump was installed for the filter, so that a higher pressure in the filter could be reached and it was possible to enhance the flow rate. To avoid defects at the pump, a separator for coarse particles was installed in series to the pump. The pump used behind the filter unit was taken out of operation. The line of the old pump is used as a new by-pass line. Filter 2 showed good results and also due to economic aspects, it was planned to substitute the precoated filter (Filter 1) by a reversible flow filter (Filter 3), too.

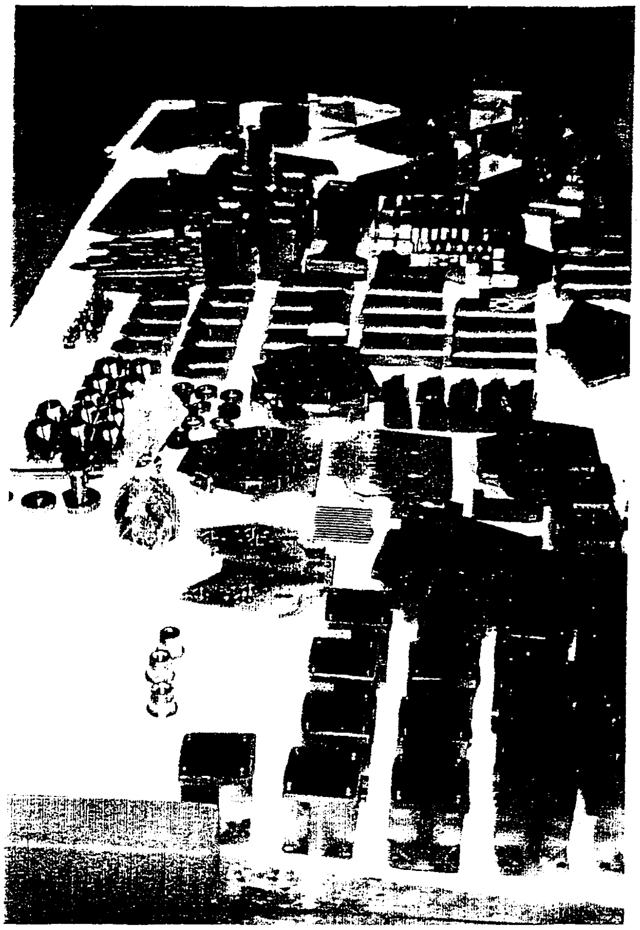


PICTURE 19 - One of the three gripping clamps for the handling of plates or flanges.



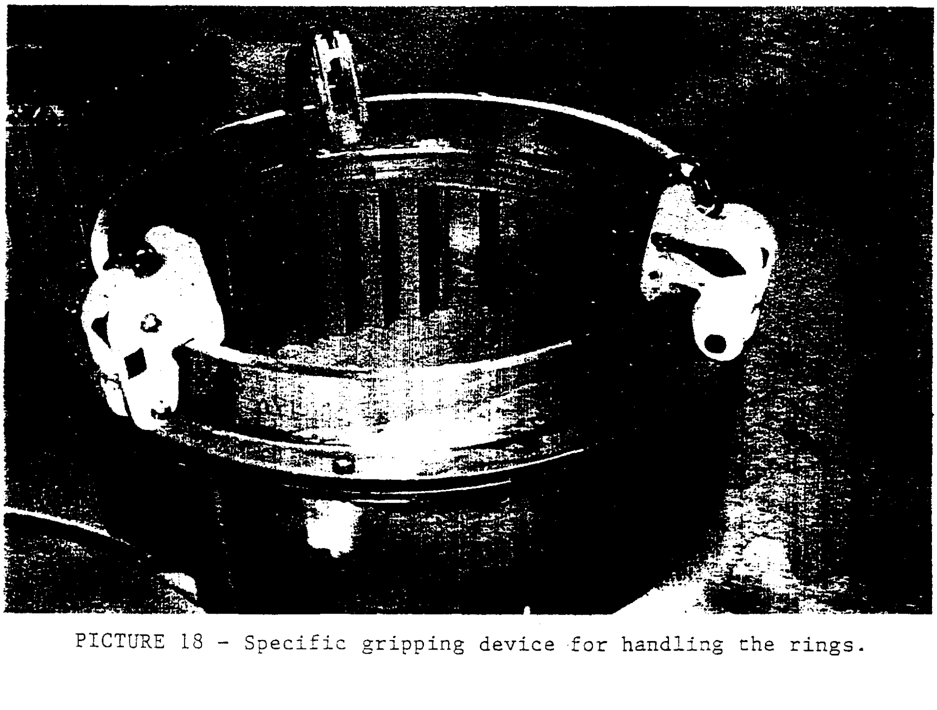
PICTURE 20

Withdrawal from the lead shield out of the refuelling canal.

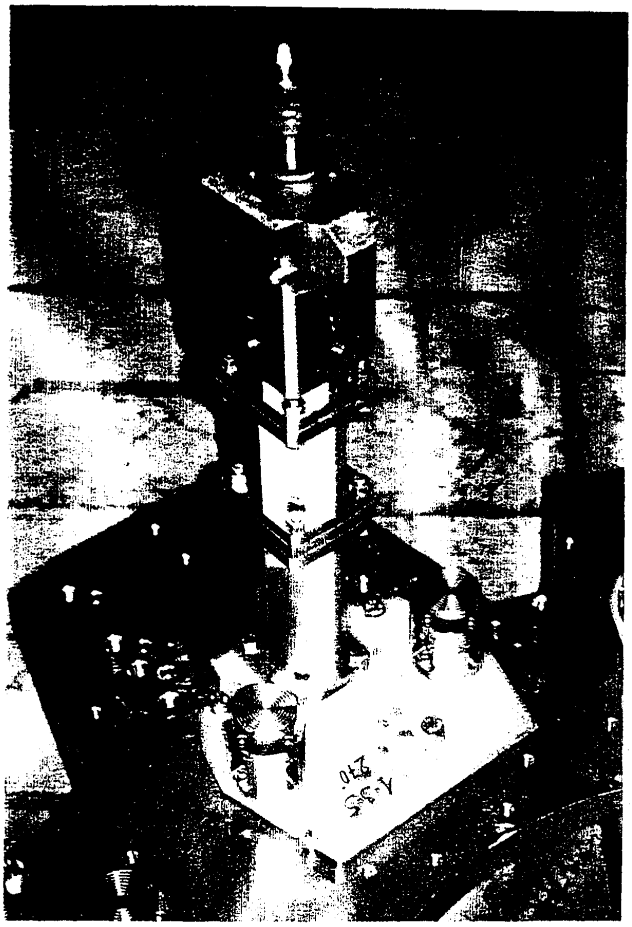


PICTURE 17

Part of the modular pieces to clamp the 11 rings.

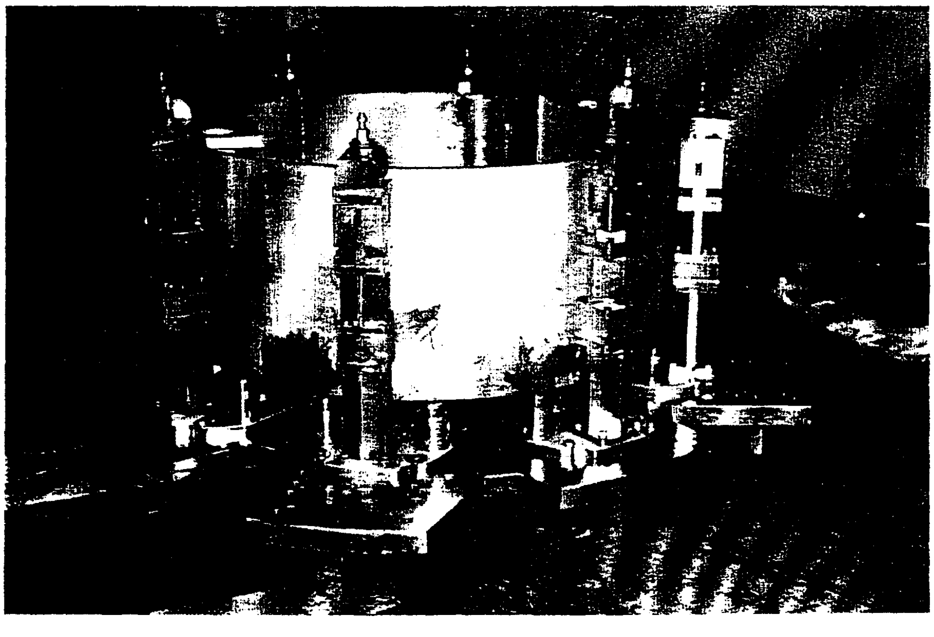


PICTURE 18 - Specific gripping device for handling the rings.

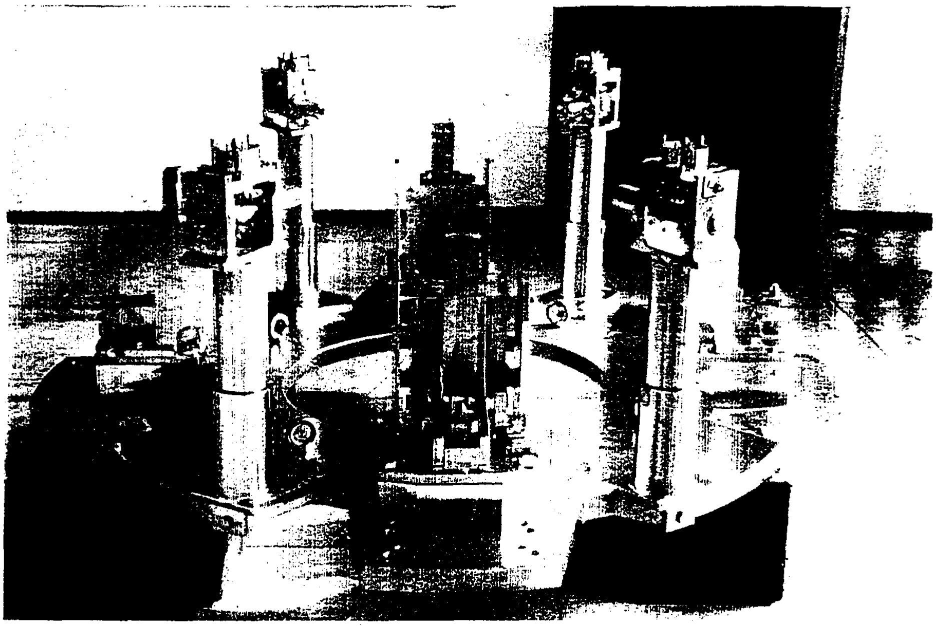


PICTURE 15

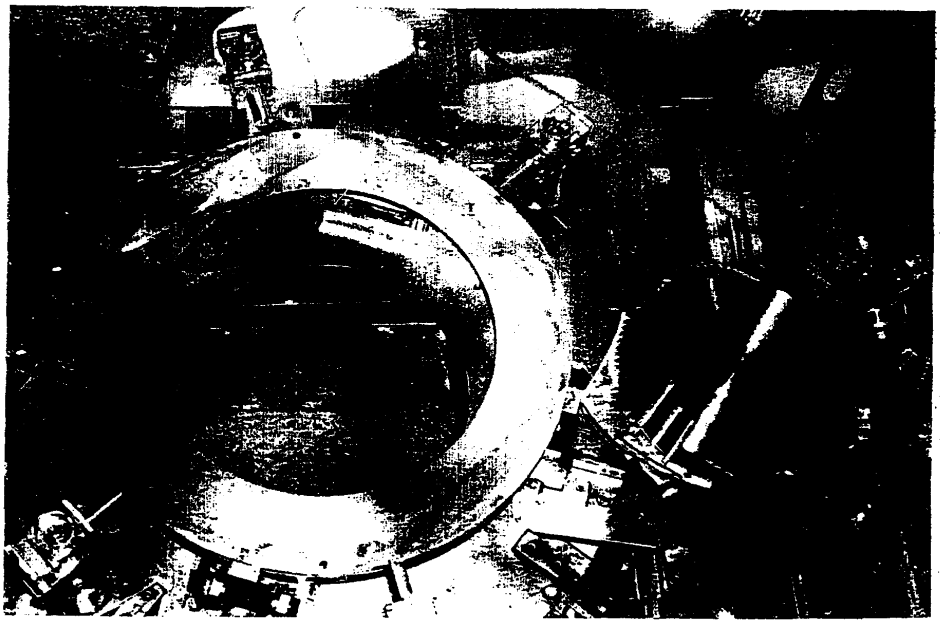
View of the clamping device unit in Step 3 configuration of the turn-table.



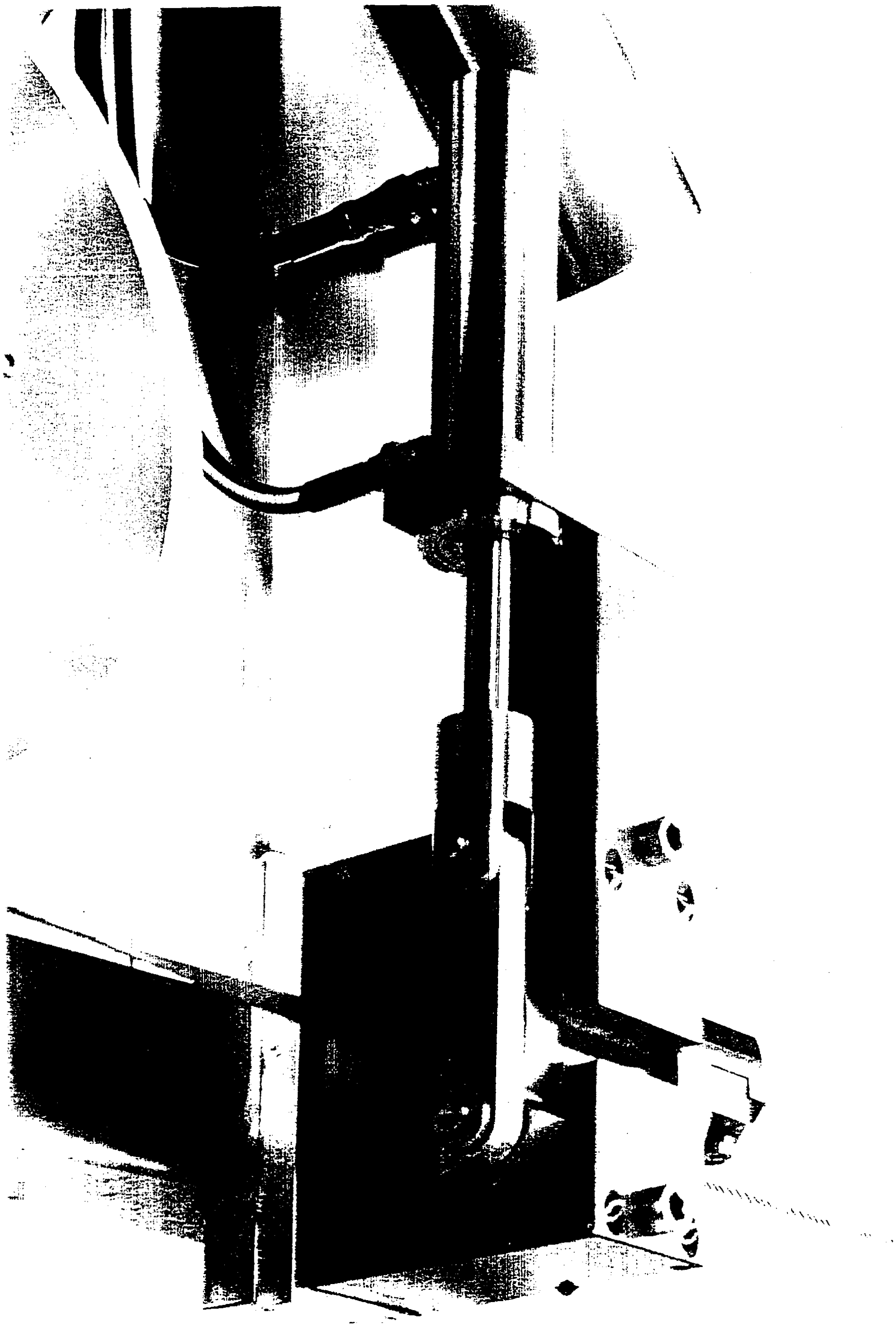
PICTURE 16 - General view of the workpiece clamped in Step 3 configuration of the turn-table.



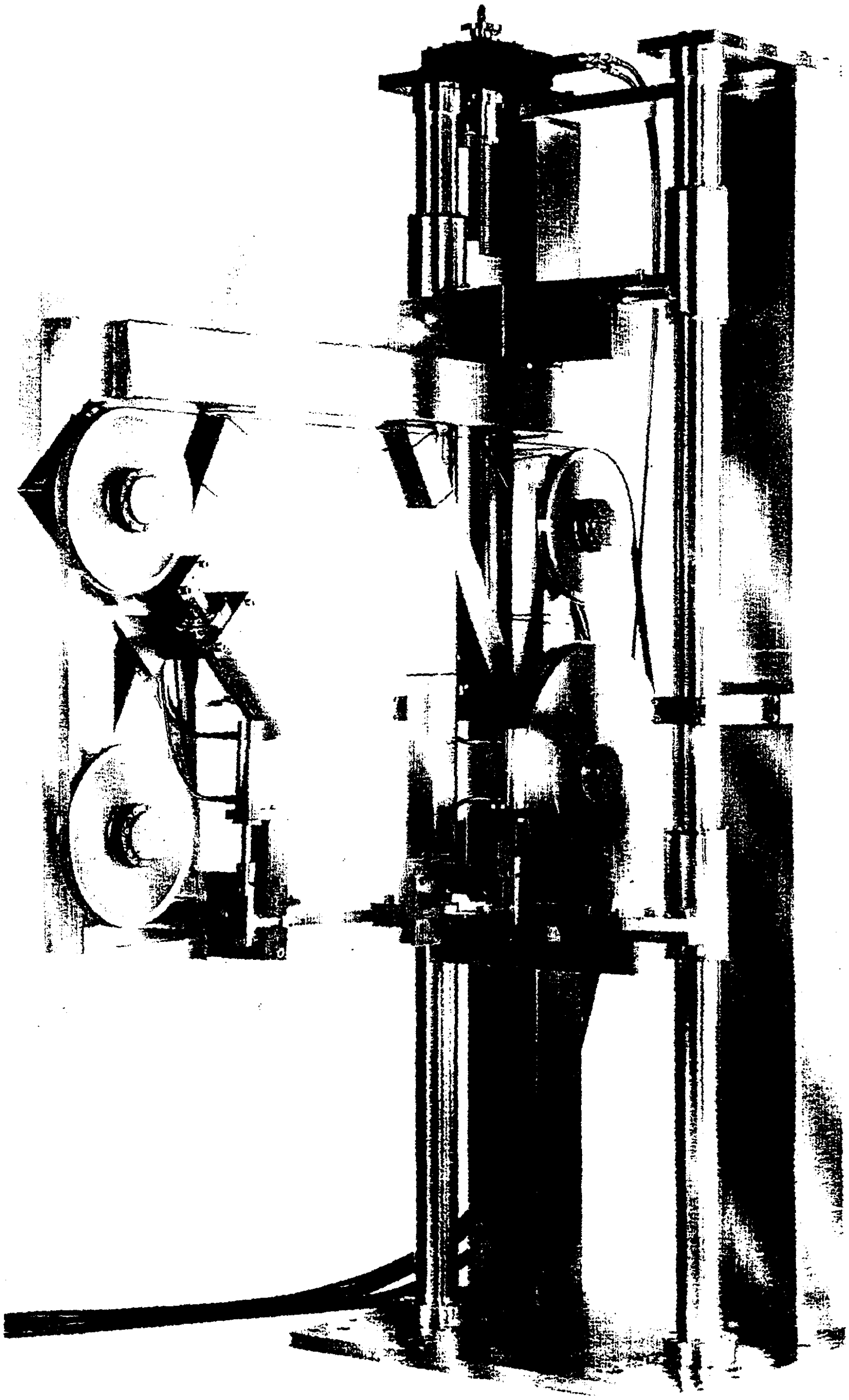
PICTURE 13 - Configuration of the additional modular plate foreseen for horizontal cut no.1 of the LCSA.



PICTURE 14 - Turn-table in Step 2 configuration during the circular sawing test.



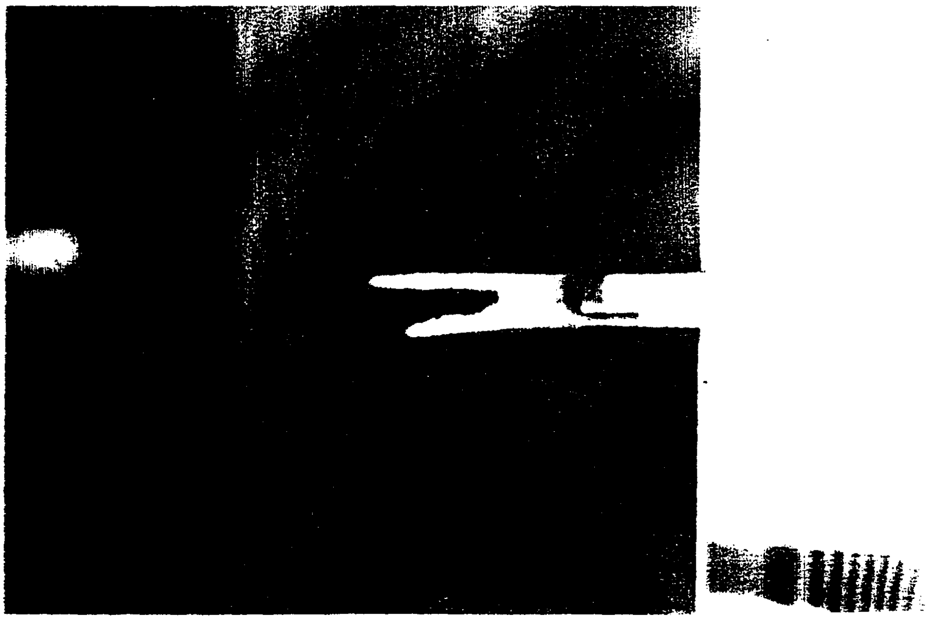
PICTURE 12 - View of 1 blade guide assembly
of the band saw

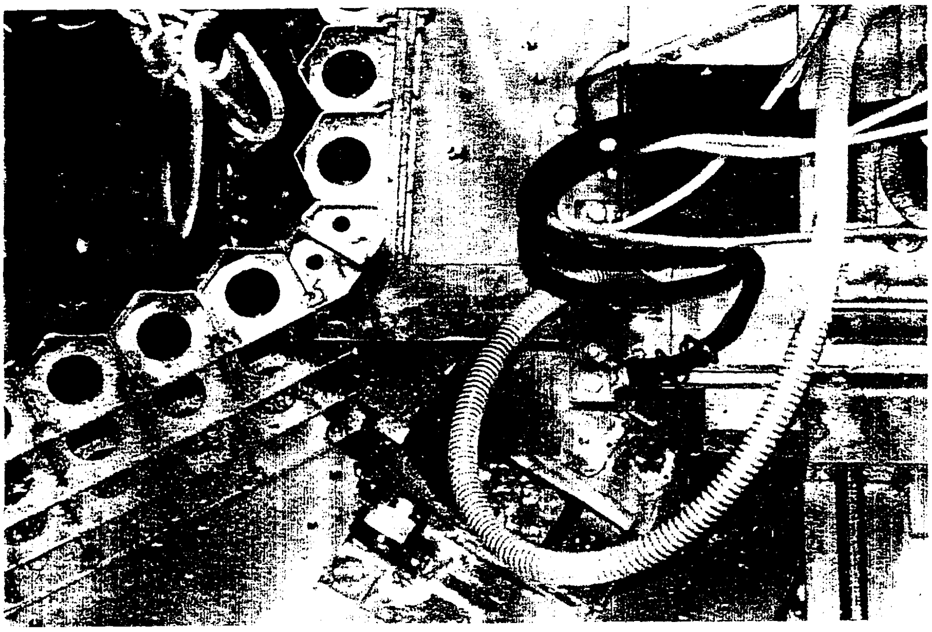


PICTURE 11 - General view of the band saw machine

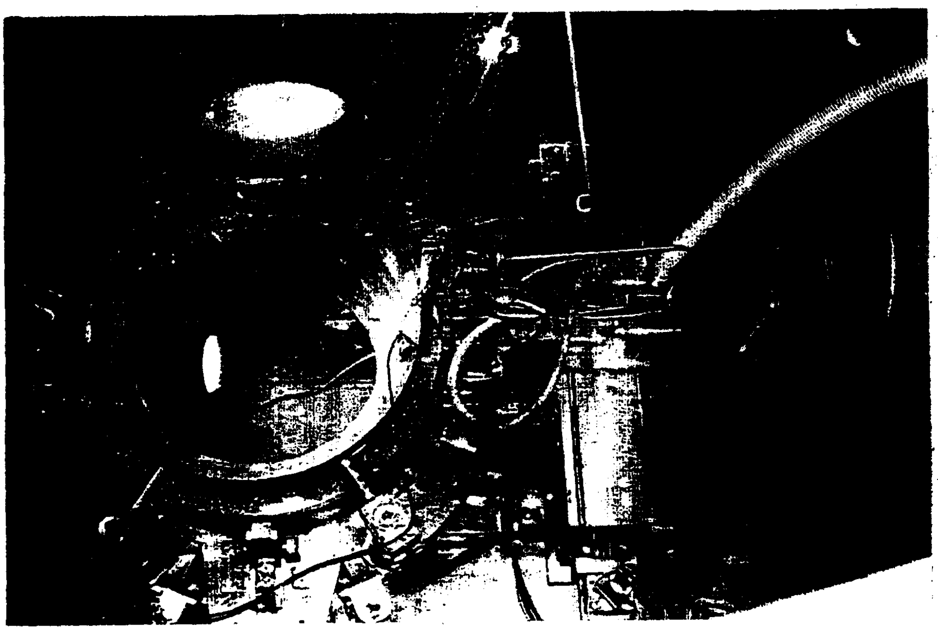
PICTURE 9 - PICTURE 10

Damaged electrode (above) : material of the core baffle (below) has entered the suction opening and opened it.

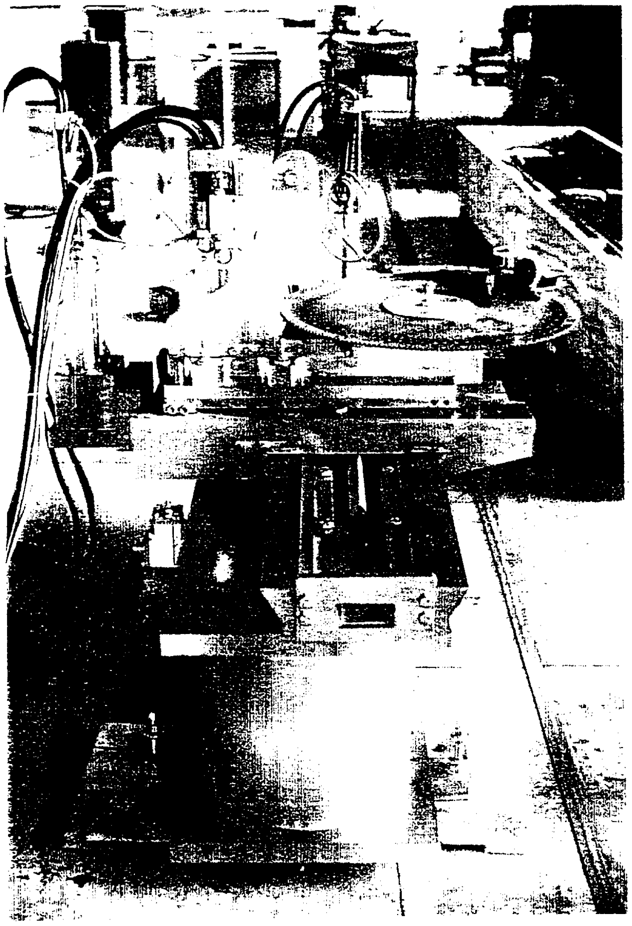




PICTURE 7 - Cutting of the core baffle by metal disintegration machining.

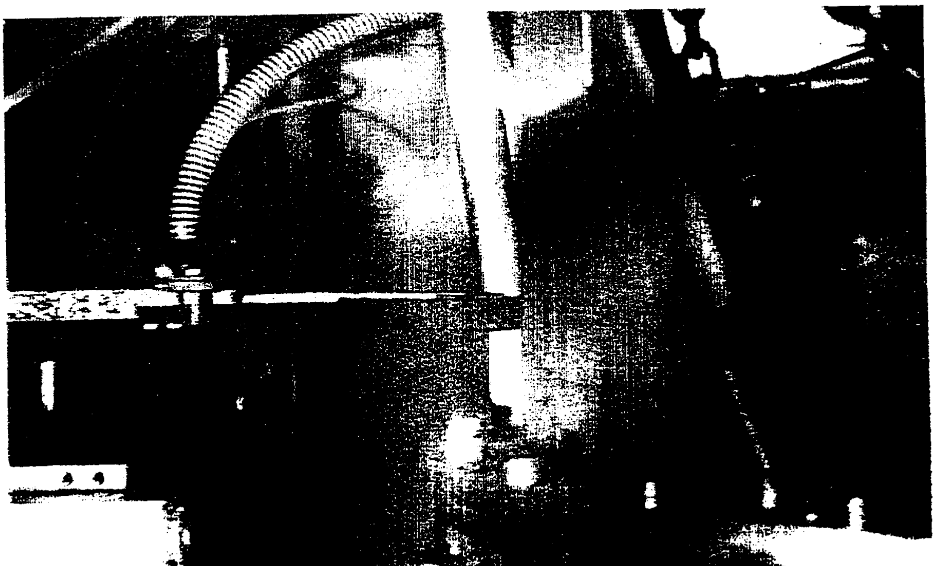


PICTURE 8 - General view of the metal disintegration machining cold tests on the lower core support assembly mock-up.

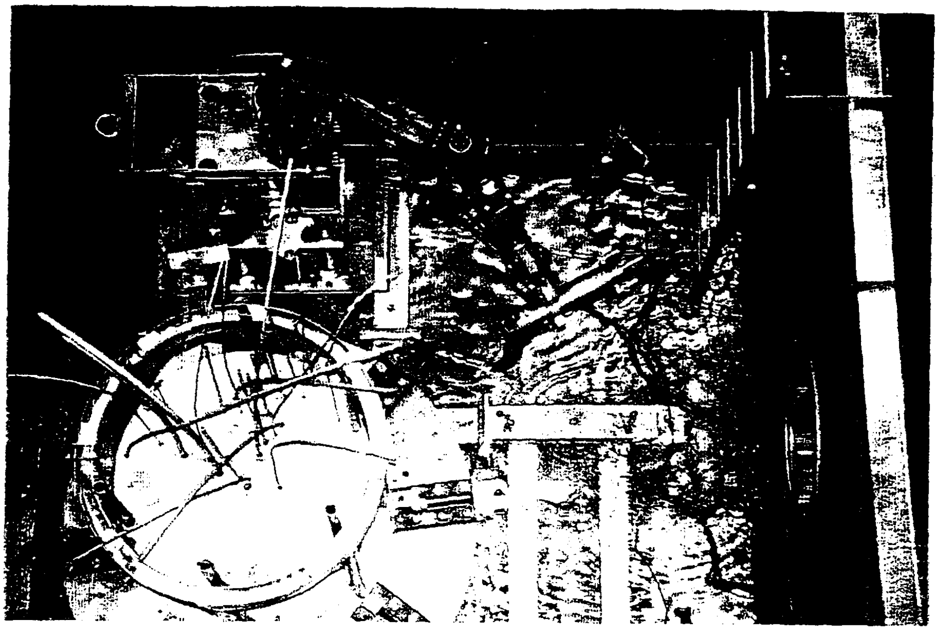


PICTURE 5

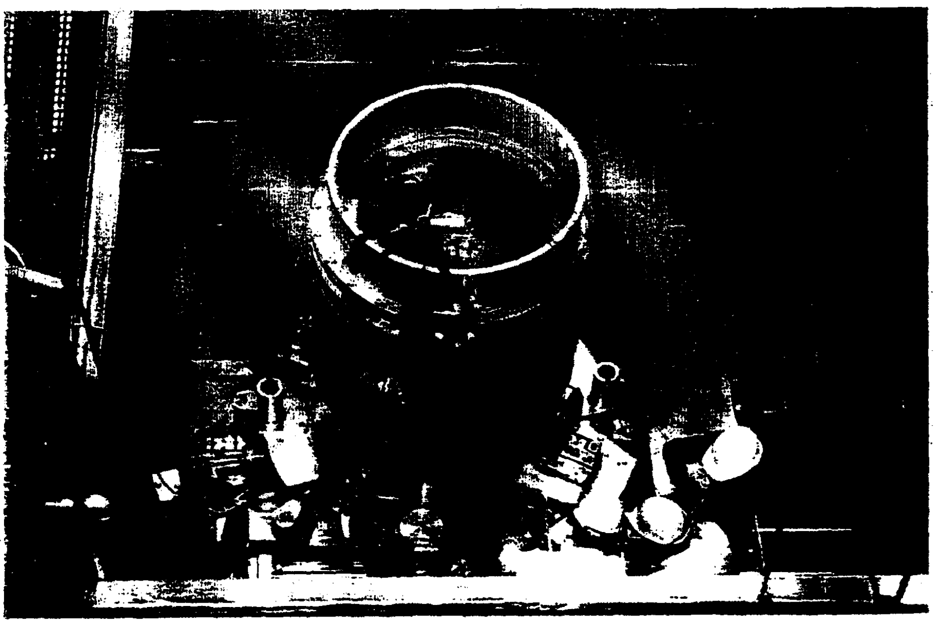
The mechanical cutting equipment consisting of a base, the X and Y positioning guides and the mechanical cutting head. The Ø 600 mm saw blade is mounted. The position of the rotary metal brush, the suction connection and the soft brushes are as delivered. They have since been modified to improve swarf collection.



PICTURE 6 - Cutting of tubes on the lower core support assembly mock-up by metal disintegration machining.



PICTURE 3 - View of the telemanipulator with the hydraulic jaw cutter, working on the upper internals mock-up. These are clamped on the turn-table Step 1.



PICTURE 4 - The lower core support assembly (LCSA) mock-up on the turn-table in the test tank, positioned for cutting at level I with a Ø 600 mm saw blade.

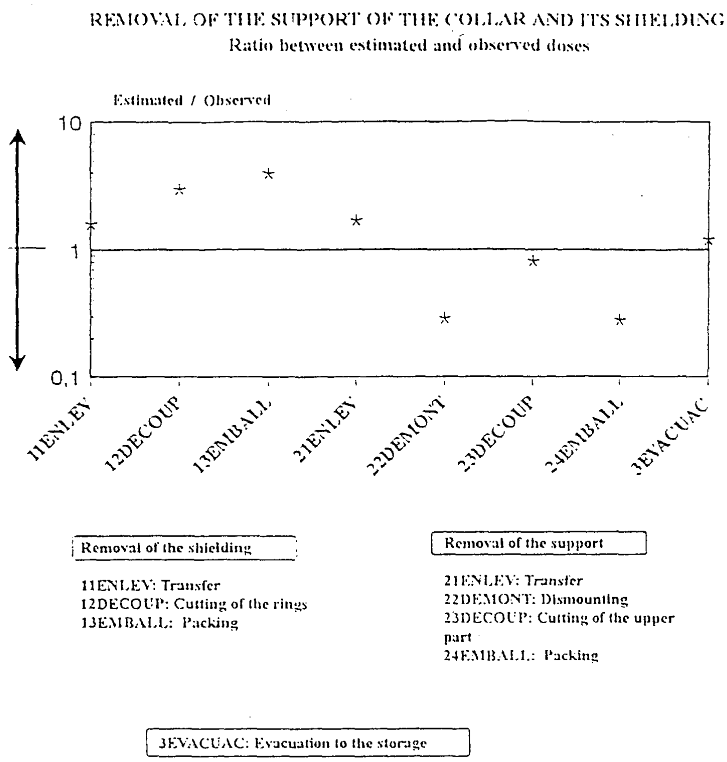
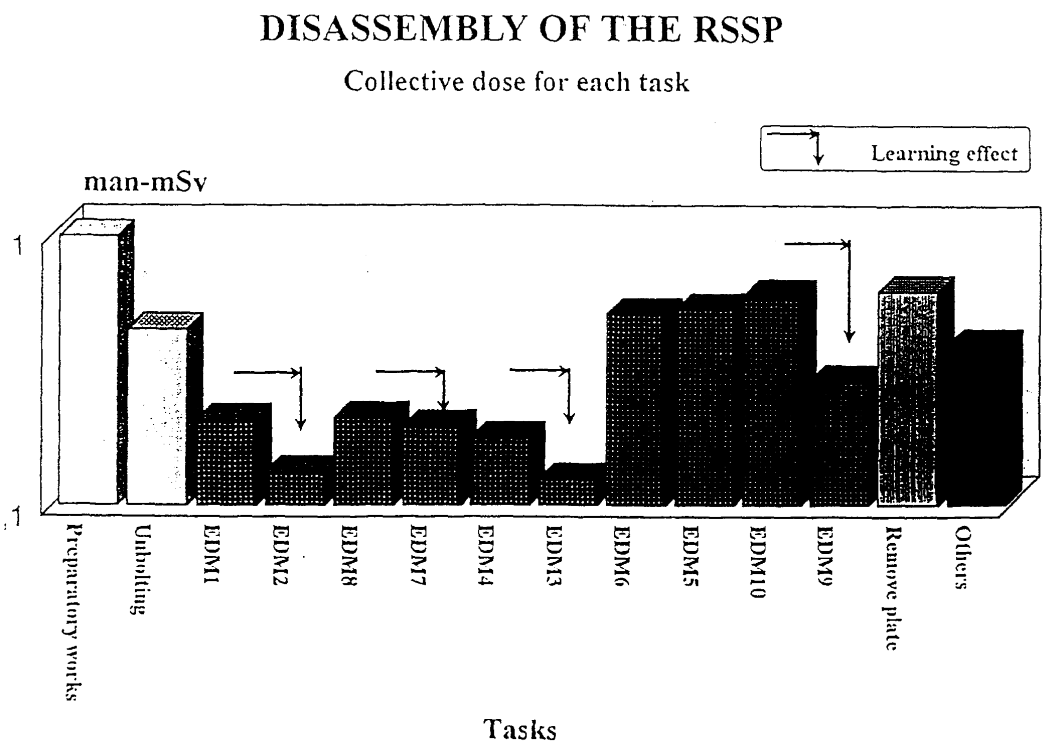


FIGURE 10

Removal of the support of the collar and its shielding.

FIGURE 11



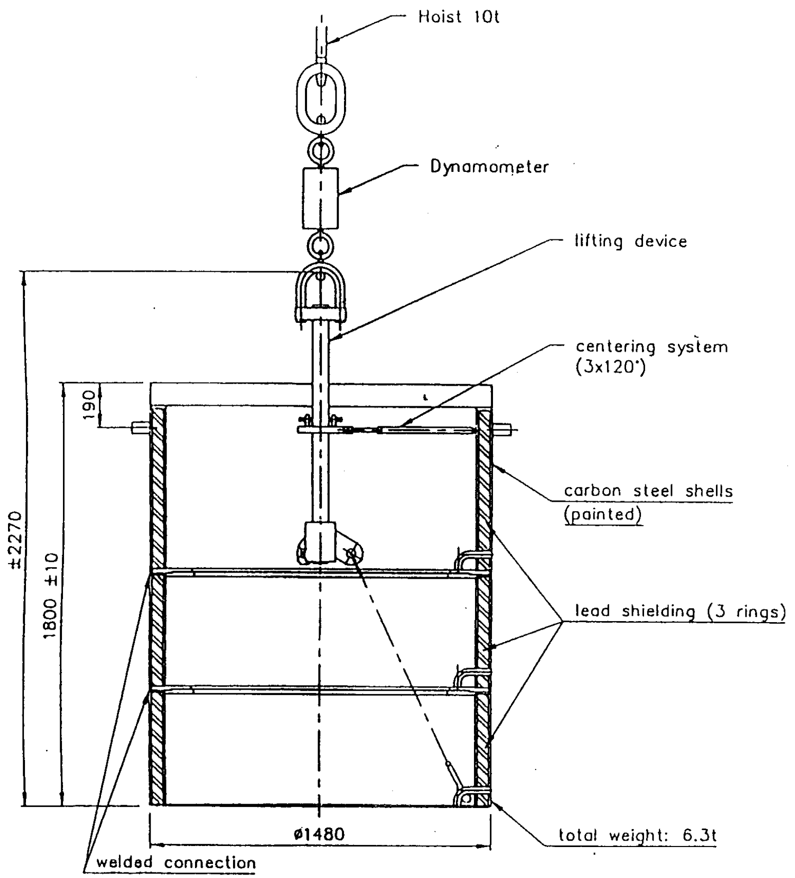


FIGURE 8

Evacuation of the lead shielding (3 rings)

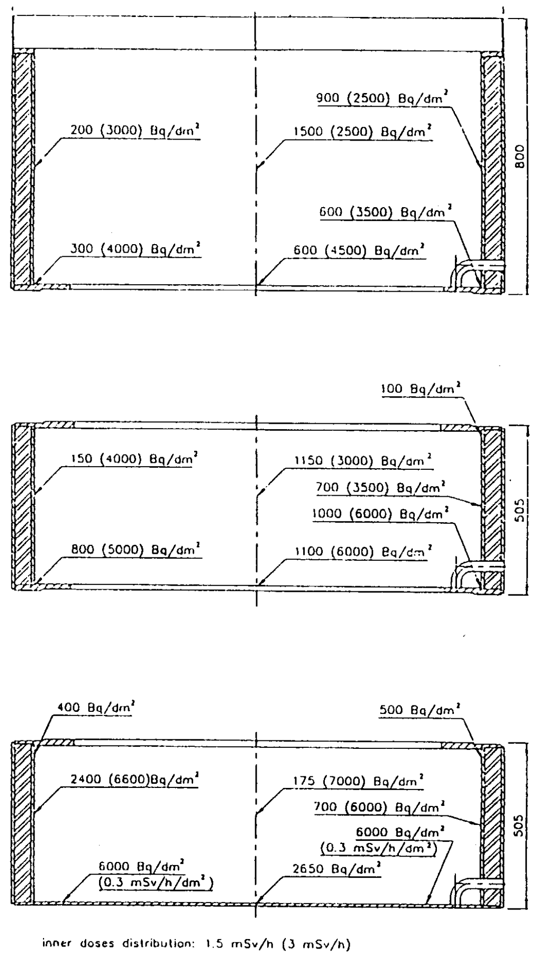


FIGURE 9

Contamination and doses distribution after (before) cleaning by flushing and high pressure waterjet

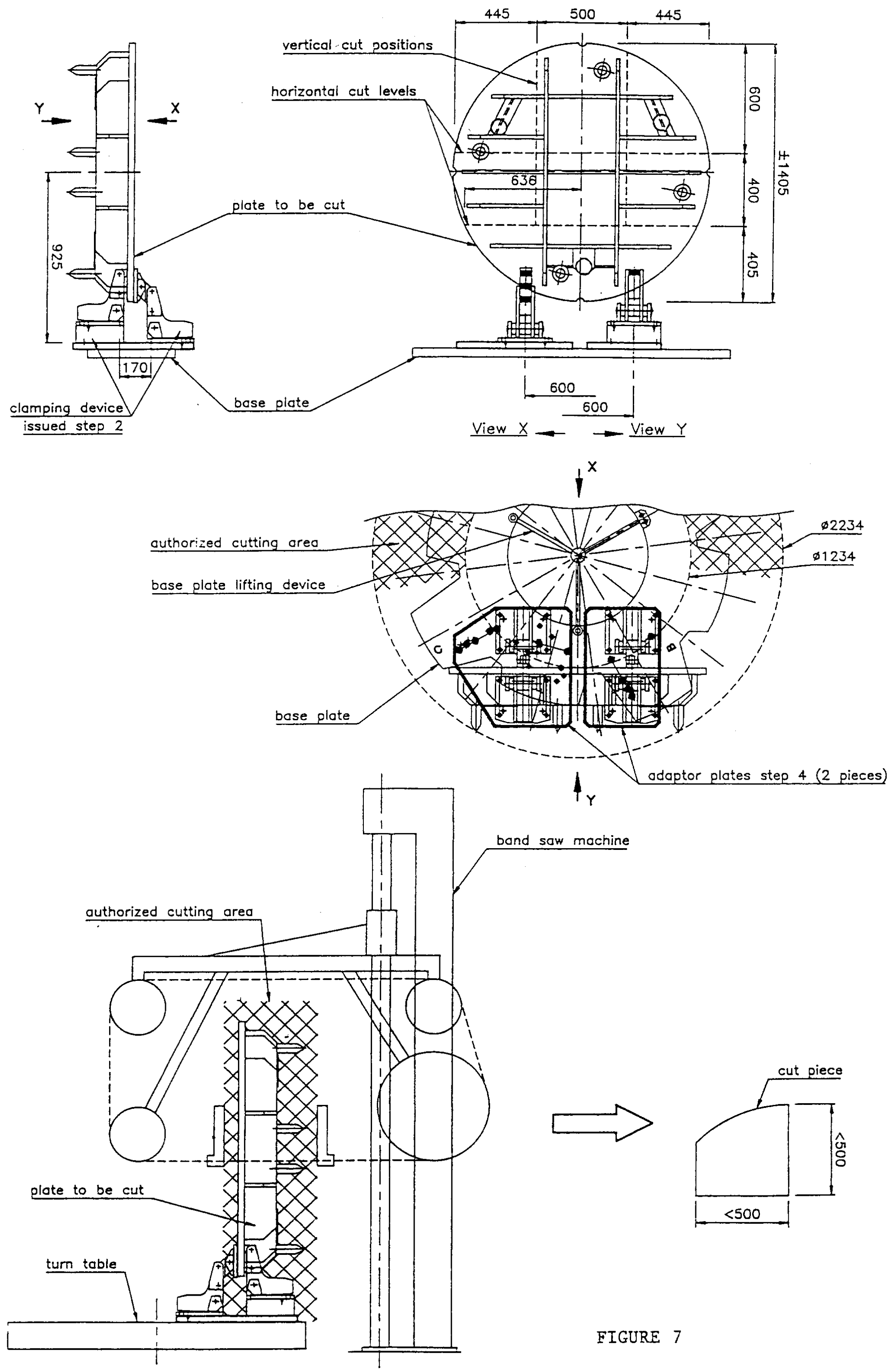


FIGURE 7

Schematic view of the step 4 of the turntable
Clamping of the plates

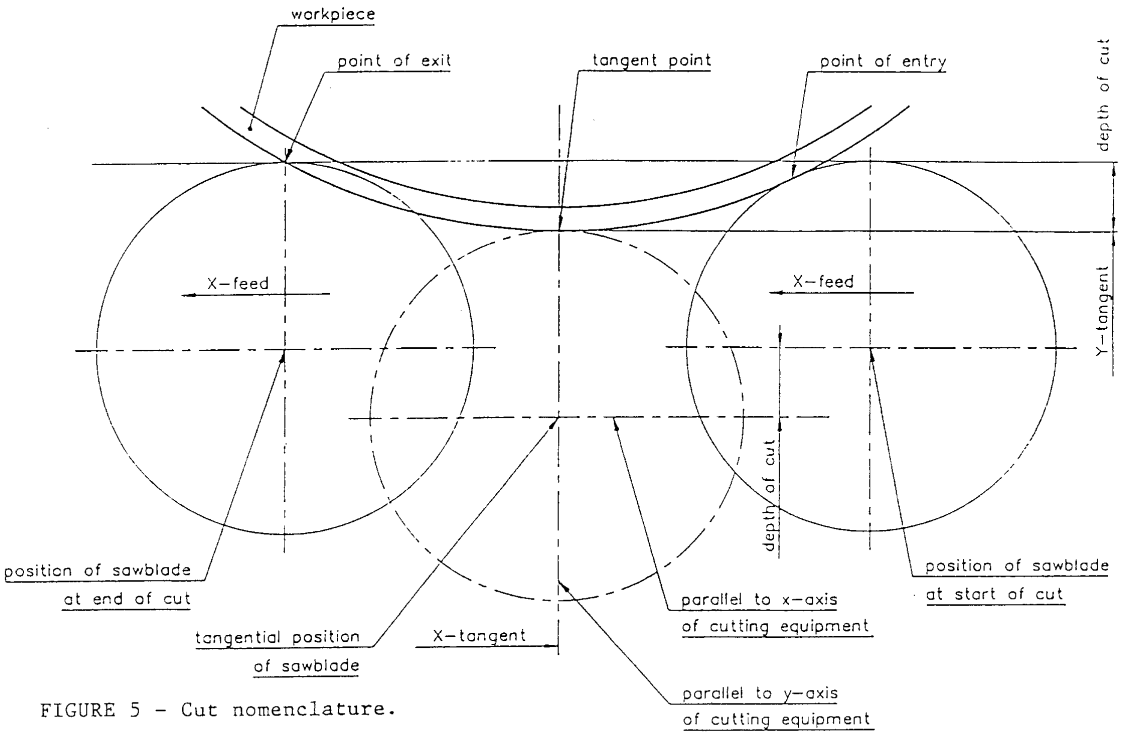


FIGURE 5 - Cut nomenclature.

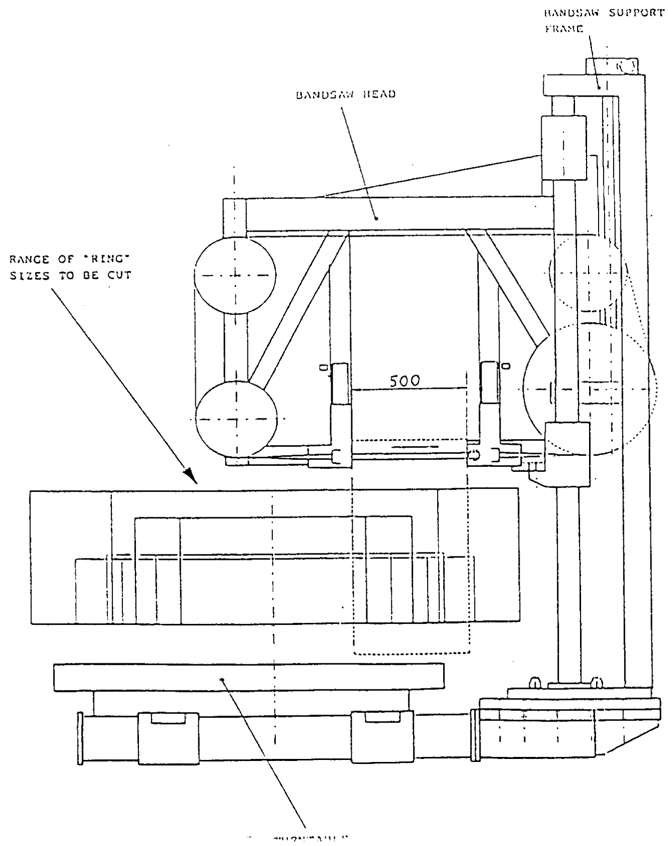


FIGURE 6

Overall schematic view
of the band saw machine.

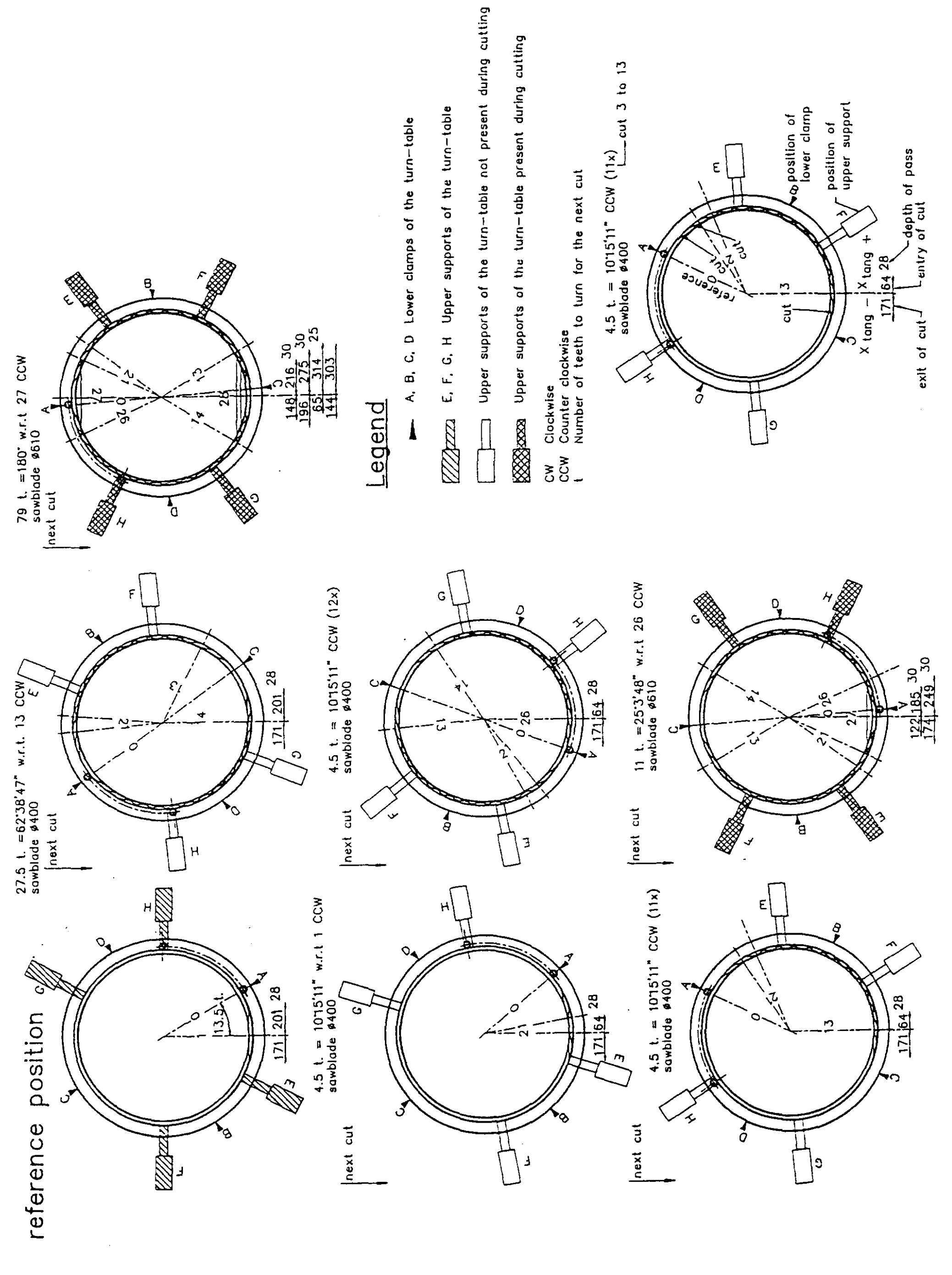


FIGURE 4 - Example of a cutting plan : level III of the lower core support assembly mock-up.

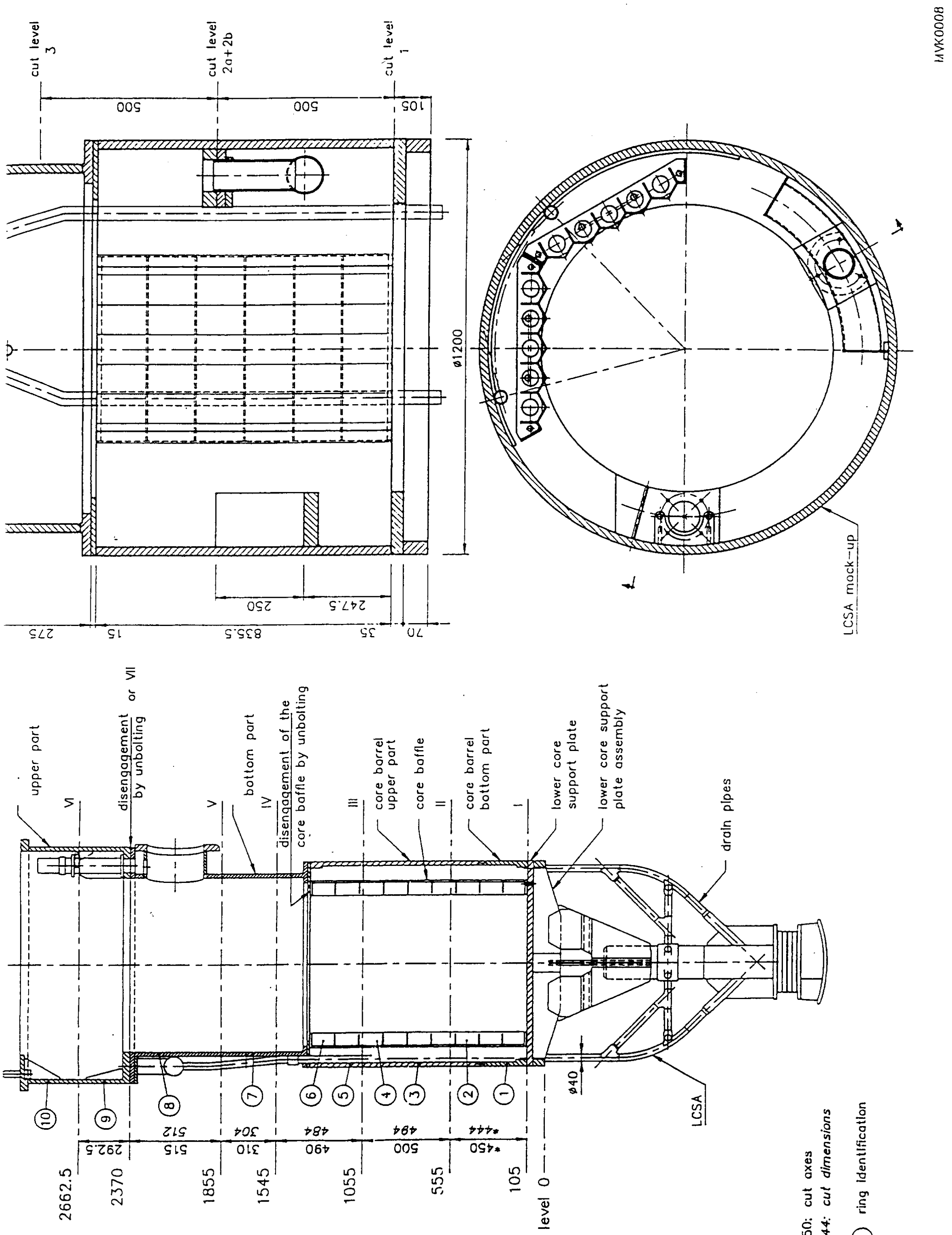


FIGURE 3 - Cutting levels on the lower core support assembly and its mock-up.

FIGURE 1 - EDM Cutting. Filters utilization.

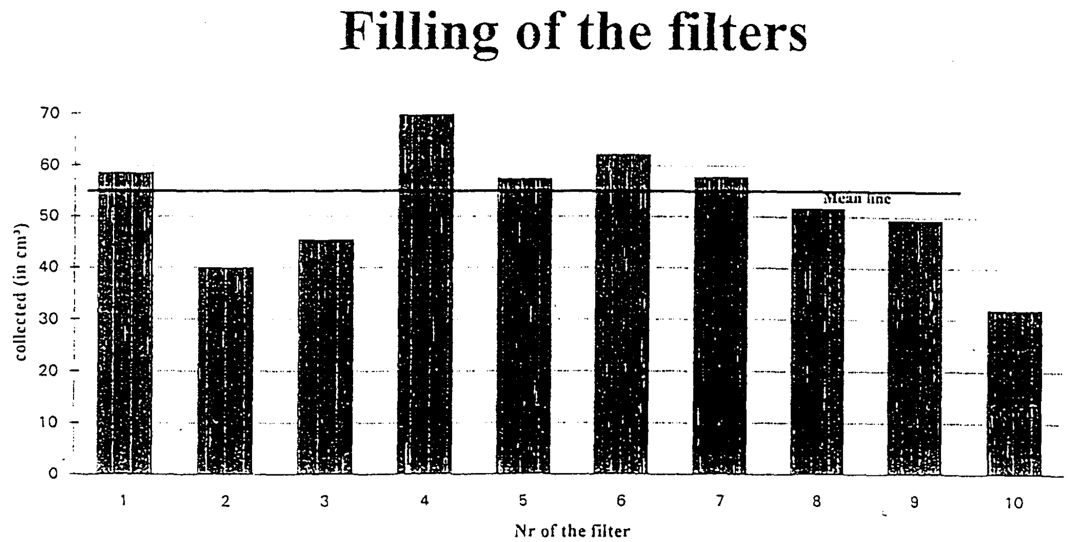
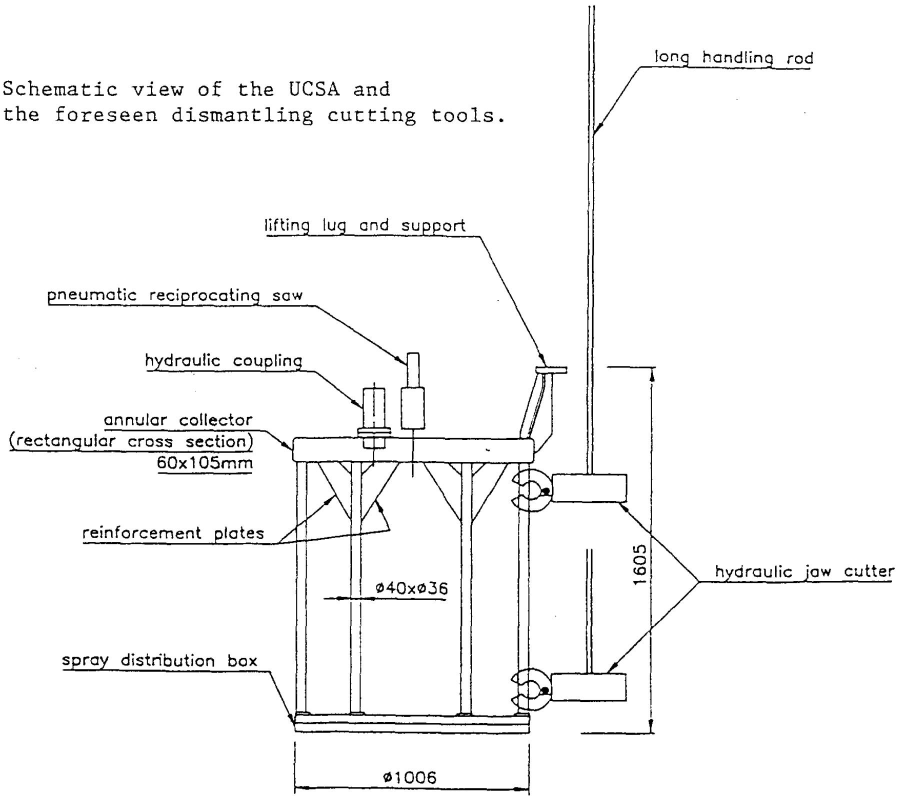


FIGURE 2 - Schematic view of the UCSA and the foreseen dismantling cutting tools.



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CEC contract F12D-0003B(TT)av.1 - 7th Progress Report

[2] The BR3 pressurized water reactor pilot dismantling project
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[3] Guide des usinages par électro-érosion et par électrochimie
Edité par le CETIM (France) et l'ENIMS (URSS)

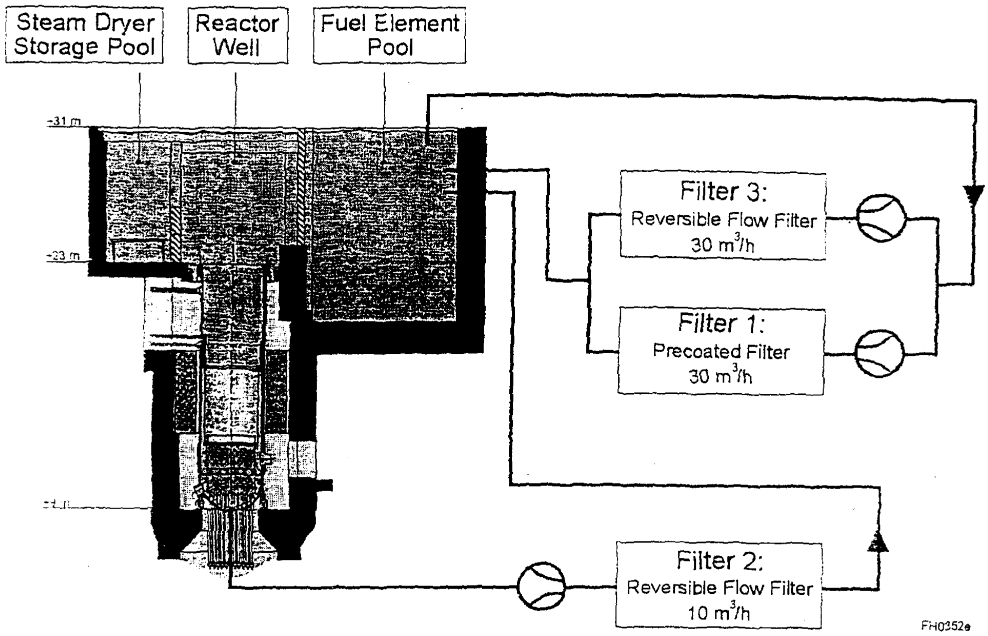


Figure 20: Advanced cleaning system for reactor water

After the dismantling of the upper grid plate and the water separator cyclones was finished in November 1998, these parts and the inner walls of the primary steam and feed water pipes were cleaned with high pressure water jet. At the same time a gradual beginning turbidity of the reactor water could be observed. It was discovered that the existing filter systems were not able to cope with this problem, especially as Filter 2 was shut down for repair measures. Filter 1 could only be supported by two mobile under water filters, which were immediately blocked. Soon the water quality became so bad, that all dismantling activities under water had to be stopped.

It was suspected, that micro-organism caused the turbidity. Since March 1999 disinfectants and oxidising agents like hydrogen peroxide were added to counteract the influence of the bacteria attack. A laboratory test showed that a pore size of 0,45 µm would be necessary for extraction of the oxidized bacteria. Due to the large volume, it was decided to exchange the water part by part for dilution. After nearly 3000 m^3 of water had been exchanged, the quality was acceptable again and cleaning of the pools from the sedimented slag was initiated.
During the period of cleaning a new turbidity of the reactor water could be observed again.

The following systematic examination of this new turbidity showed, that the bacteria pollution could be filtered through filter 1, by precoating it with activated carbon. But the cleared water had to be stabilized. This was realized by treating the reactor water with ozone. Other suspended particles could be separated with the mechanical filters. Due to the organic concentration it was not efficient to clear the water by filtration only. Therefore the exchange of water started again to reduce the total amount of dirt. Since end of 1999 the water quality is good enough, to restart the dismantling work under water again. The additionally installed reversible flow filter (Filter 3) has taken up its work this time and seems to guarantee a good water quality (**Figure 21**).

As a consequence of this water turbidity handling and working under water was completely interrupted, which caused a delay of the whole project of more than one year.

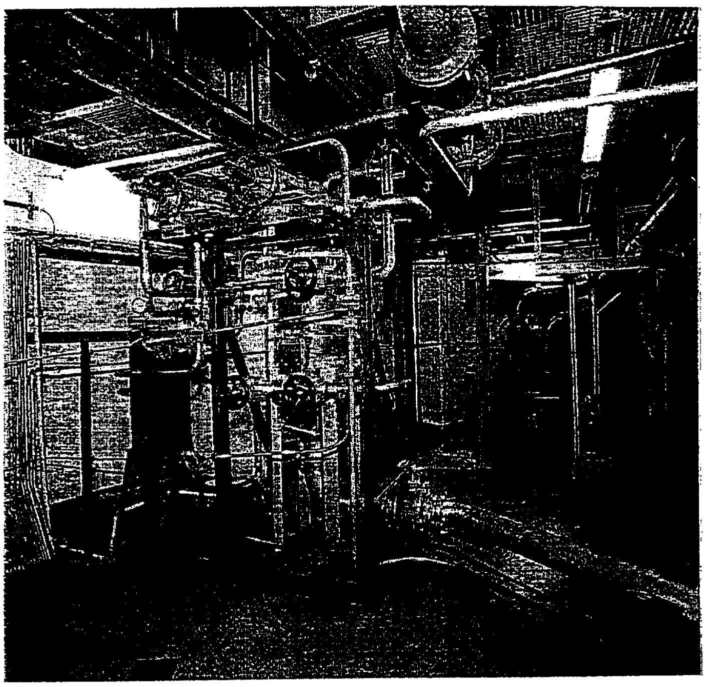


Figure 21: The new reversible flow filter

It may be stated for the interests of decommissioning in generally, that the consequence of dismantling highly activated components with the subsequent removal of radioactivity from the reactor pool should be taken into account in view of water purification problems. It is recommend to provide for bacteria pollution by treating the water right in time, e.g. with ozone. With this measure the missing effect of the removed radioactivity can be compensated.

B.2.4. Full scale testing of techniques

BR3

One of the most important lessons learned during former decommissioning experience is that the cold testing of techniques on mock-up can avoid dose uptake and cost due to the unavoidable learning process and the definition of cutting parameters in an easier situation. So all techniques using engineered machines were primarily cold tested.

Five such operations were identified. We refer here to the global decommissioning strategy extensively explained in B.4 below. It concerns:

- the internal pipe cutter in order to desolidarize the RPV from the hot and the cold legs near the RPV;
- the reinstallation of the water tightness of the NST and the reactor pool;
- the removal of the insulation shroud and its fastening profiles;
- the horizontal cutting of the RPV using the milling cutter machine;
- the vertical cutting of the RPV using the band saw machine.

➢ ***Internal pipe cutter***

The equipment (**figure 22**) was tested on a mock-up of the cold leg in a layout as close as possible to the real situation.

The internal pipe cutter is a prototype cutting equipment designed to cut a (very) thick pipe from its inside. The particularity is a very low ratio inside diameter/thickness (close to 2.1).

A cylindrical milling cutter is rotating around its axis (300~380 rpm) and can be moved radially inside the pipe and tangentially. The first method of cutting originally followed an alternated path with increasing radius (clockwise and counterclockwise rotations): this led to instabilities of the equipment. So the cutting method had to be adapted and the milling cutter did finally cut through the piping wall following a spiral shape.

Rotating speed (cutting speed): 300~380 rpm
Cutting depth: 15≅20 mm
So the radial feed was normally operated in order to keep the cutting depth of the tool at constant values.
Tangential feed speed: ≅ 35 mm/min.

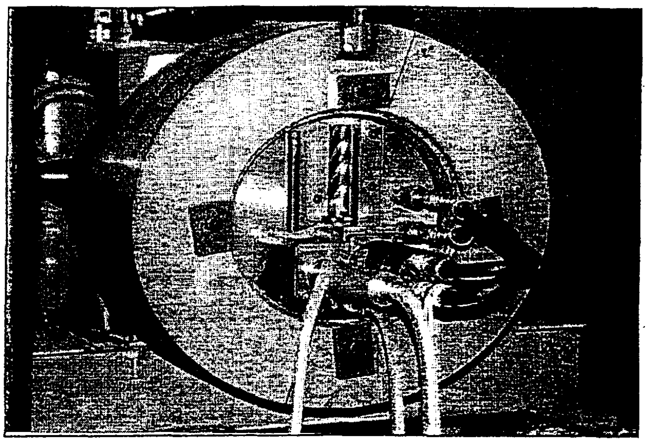


Figure 22: Internal thick pipe cutter during cold tests

The cold tests allowed to set up the cutting parameters for this cutting technology.
In addition it was possible:

- to optimize the cutting methodology and the cutting parameters;
- to control the equipment size in the pipe in the neighbourhood of the first elbow of the cold legs;
- to visualize the operations inside the pipe, to control the cutting data and behaviour of the tool;
- to study the better management of the chips produced (recuperation & evacuation).

➢ ***Reinstallation of the water tightness of the NST and the reactor pool***

The desolidarization of the RPV from the reactor pool and from the primary pipe (by cutting the legs) has destroyed the water tightness of the reactor pool. Before the lifting of the RPV, the pool must be filled again. So the water tightness must first be reinstalled. The solution found consists in placing and fixing a sealing device on the inner surface of the neutron shield tank in front of each primary pipe penetration hole.

A first test was organized in order to analyse the remote positioning as well as to control the leak tightness of the system. This first test led to the following conclusions:

- the remote positioning of the sealing piece had to be improved;
- concerning the leak tightness, irregularities in the construction of the mock-up did not allow to conclude that the system fit perfectly.

So a second test was foreseen on a mock-up with a reviewed design. This test was successful and the actual operation could be prepared (building of the 3 required sealing devices) and scheduled.

Figure 23 shows the cold test of the introduction of the new sealing device in the adapted NST mock-up.

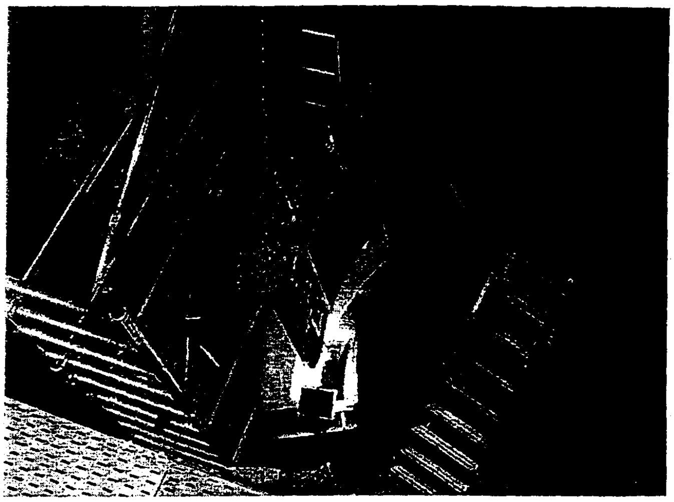


Figure 23: Introduction of the new sealing device in the adapted NST mock-up

➢ ***The removal of the insulation shroud and its fastening profiles***

This operation was carried out using a remote hole cutter to destroy the screws maintaining the insulation shell. The hole cutter is an industrial model that has been adapted to work remotely and under water. A specific supporting equipment and a mock-up were also designed and built. The full scale test was carried out in a test tank in the non-controlled area of the BR3 plant.

➢ ***The horizontal cutting of the RPV using the milling cutter machine***

The chosen dismantling strategy reuses the existing circular saw (and the existing band saw). New tests were necessary to define new machine cutting parameters because compared to the previous phase of the project (i.e. dismantling of the reactor internals), a different sort of base material (carbon steel instead of stainless steel) and another piece thickness (112 mm instead of 25 mm) had to be cut.

Due to the fact that the cutting equipment was contaminated during previous phases of the project, the cold test had to be carried out in the controlled area.

For the circular saw, a new clamping device was designed mainly for the first cut. The purpose is to clamp the upper part of the RPV during and after the cut. The lower part of the RPV is difficult to clamp due to its spherical shape (**see figure 24**).

A mock-up of the reactor pressure vessel was made to carry out tests of the cutting technique with the circular saw.

A first series of tests brought some problems to light:

- the mock-up vibrated a lot;
- the type of sawblade seemed not ideal for the purpose (the metal chips looked more like iron filings with a length smaller than 3 mm).

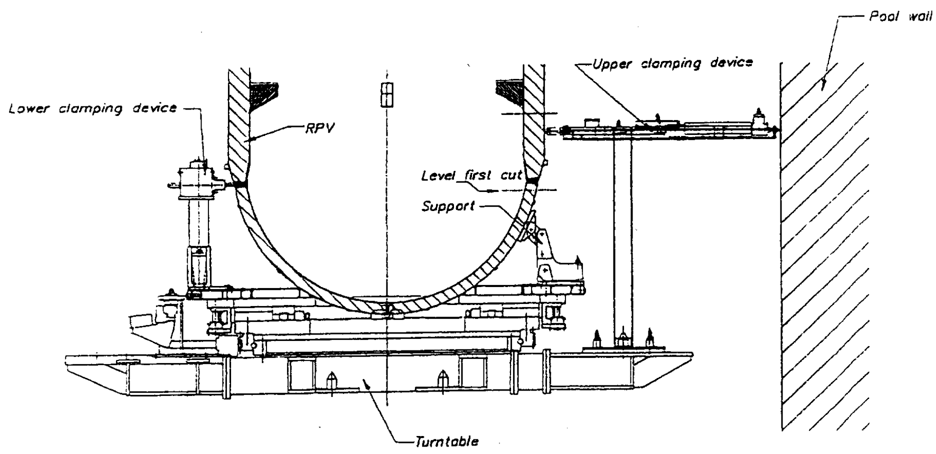


Figure 24: The clamping system for the first horizontal cut. The cold tests showed that the system was too weak to resist the involved cutting forces.

The results of this first series of tests led to:

- the construction of a stiffer clamping system;
- the detailed analysis of the requirements of the cut, concerning the saw blade design;
- the organization of an extended second series of tests.

The second series of tests had as objectives:

- to validate the new clamping system device (see figure 25);
- to try and optimize the cutting sequences and their associated parameters.

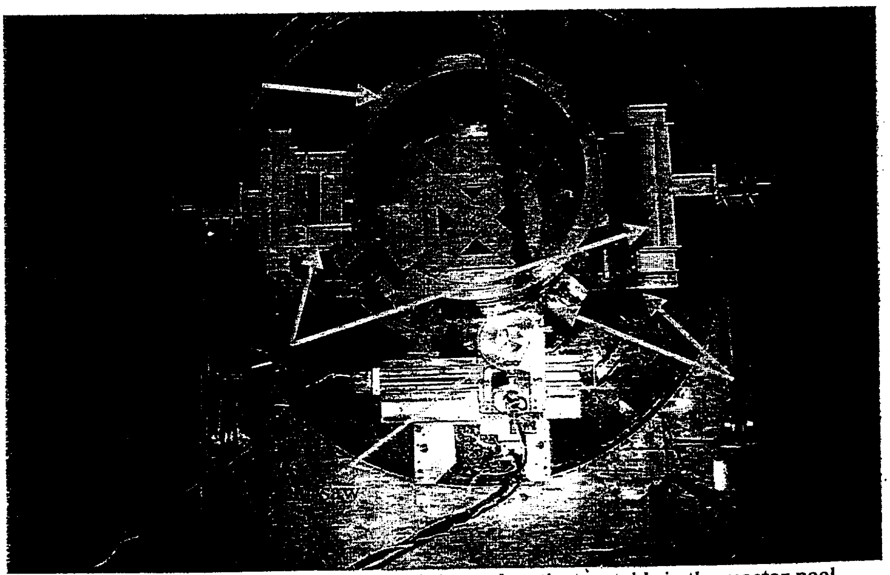


Figure 25: Mock up of the RPV clamped on the turntable in the reactor pool using the new clamping system

Both objectives were reached and the selected saw blade type had the following characteristics:

- outside diameter: 500 mm
- tooth pitch: 21 mm
- tooth thickness: 6 mm
- toothing: according DIN 1838 (pre cutting teeth about half the thickness but higher by ~0.3 mm than the finishing teeth).

Even with an eccentricity of about 0.8 mm (which is much too large), the generated loads during the cutting tests remained at an acceptable level. The optimized parameters were:

- tangential speed: more than 15m/min (rotation at about 10 rpm)
- cutting depth: up to 30 mm
- feed speed: 15~20 mm
- chips length: 10 mm as minimum.

The eccentricity of the saw blades made that only a quarter of the circumference (and thus of the total amount of teeth) of the saw blades were actually used.
The optimized parameters were used to evaluate the actual job but it was immediately clear that an important improvement could be reached by giving more attention to the eccentricity. A smaller eccentricity would lead to better parameters and an extended lifetime of the blades as well. Therefore, saw blades with improved eccentricity were ordered for the actual RPV cutting.

➢ ***The vertical cutting of the RPV using the band saw machine***

The foreword mentioned for the milling cutter is also valid for the band saw.

Vertical cuts through the nominal thickness of the RPV (112 mm)

A few cuts were carried out in the SS cladded carbon steel wall of the RPV mock-up without any problem using a feed speed of about 20 mm/min. This immediate success is thanks to a similar job that the BR3 team already carried out during the previous phase of the project (i.e. the dismantling of the instrumentation collar).

Vertical cuts directly through the RPV flange followed by a horizontal cut through vessel-insulation-shroud (all 3 in the same time)

The cuts through the flange (thickness 355 mm) were successful at 7 mm/min. Special attention must be paid to the end of the cut of the weld situated below the flange (risk of blade jam).
The major difficulty for the cutting of the RPV flange is that there is a supporting shroud (thickness ¾") under the flange. This means that for cutting the upper part of the RPV, the band saw has to cut in the same time the flange followed by the vessel + insulation + shroud. The influence of the insulation on the cutting performances was unknown and had to be tested. Finally, these cuts were carried out without any problem.

EWN

The major aims of the model dismantling are the testing of the tools and working processes to guarantee the adherence to the protection goals for the later cutting of the active components as well as the technical feasibility. By an extensive test programme in the presence of the authorised expert it will be proved that the used tools are suitable and a technical optimisation of the individual steps can be achieved. Special emphasis is given to the expected radiation exposure of the personnel, the maximum application time of the tools and the minimisation of the secondary waste.

Approach for the realisation of the aims

To test the equipment for the cutting of the reactors 1 to 4 as realistically as possible, the original cutting technique will be tested with the non-contaminated original components of reactor 7 (core basket, cavity bottom, reactor cavity, protecting tube unit) and reactor 8 (reactor pressure vessel) in the low-contaminated steam generator room of unit 5. The cutting of the reactor pressure vessel nozzles will be performed with the reactor pressure vessel in unit 7 in installation position. This pressure vessel is also not contaminated. Therefore, a newly developed milling device is foreseen to be used. The cutting of the connection nozzles is a necessary precondition to lift the reactor pressure vessel out of its installation position and to transport it to the dry cutting place.

At a later stage, the cutting of the annular water tank (including reactor insulation) with a wire saw will be tested on a model.

For the installation of the equipment and the execution of model dismantling, a separate licence was applied for. This was granted in May 1997.

The results of model dismantling will facilitate the granting of the licence for cutting the activated reactor components of the units 1 to 4.

Test programme structure

The test programme for model dismantling is hierarchically structured and includes the following levels:

- test complexes
- test measures
- test processes
- test steps.

There are the following 10 test complexes:

- transport to the cutting places
- cutting and handling technique for reactor pressure vessel
- cutting and handling technique for core basket
- cutting and handling technique for reactor cavity with cavity bottom
- cutting and handling technique for protecting tube unit
- operation of packing station and container handling
- operation of support and auxiliary systems
- maintenance
- sampling RPV units 1 to 4 (EU-project)
- dismantling of RPV insulation and annular water tank.

These test complexes are divided into 44 test measures with 274 test processes and approx. 2500 individual test steps.

On the level of test processes, corresponding test plans will be elaborated for each process. Before testing, the authorised experts will announce in which test steps they want to participate. The test plans are the basis for the execution of the individual test steps and include the following:

- further subdivision of the test process into test steps,
- description of main items and goals of the test steps,
- reference to basic work documents (also drawing),
- description of results,
- personnel registration, time registration,
- personnel and training evidence.

After realisation of each test process (test plan), a corresponding test report will be prepared. The test report evaluates the test processes. The aim of the test report is to give the evidence that the technological processes with personnel participation under active conditions are performed under adherence of the limit and guide values of radiation protection and the facilities and equipment are qualified for the work.

The test report includes the following information about each test step:

- information about dose expectation on the basis of the calculated ambient dose rate at the working places of the activated units 1 to 4 and information on the efforts for the realisation of the test steps during model dismantling,
- summary information on the qualification of the newly installed facilities and equipment.

The test report will be prepared by EWN and following confirmed by the authorised experts. On the basis of the reports for each test process, the final report will be prepared and submitted to the authorised experts for approval. This report forms the basis for the granting of the licence for cutting the reactors 1 to 4.

KRB-A

The plasma arc technique is the method-of-choice for underwater cutting in KRB A. It is used for cutting the internal components of the reactor and must be used in a water depth down to 22 m. Compared to mechanical techniques, thermal cutting tools, like plasma arc cutting, are hardly used in the field of decommissioning today. Nevertheless it is obviously an advantage to have a very powerful cutting tool, which at the same time is small and flexible.
The principle features of plasma arc cutting always seemed a likely assumption, that visibility is poor after a short time of cutting because of fine particles in the water. It was already shown during cutting of the steam dryer system, that such problems did not appear.

Another question still was the spreading of activity by aerosols, especially when cutting highly contaminated and activated components.

Prior to the first use of plasma arc cutting under water in KRB A the amount of expected aerosols were evaluated in inactive tests. First positive results were gained already in 1991 from real application during cutting the shell of the water steam dryer. The outer shell had a contamination of about 6 000 Bq/cm^2. The aerosols were captured and filtered at the water surface by a special suction device. Measurements of the aerosols showed that the concentration in the exhaust duct of the local suction device was 18 Bq ^{60}Co/m^3 at the maximum. Also no increase of the specific activity in the water could be found. Suspended particles in the water from the cutting process were filtered by the filter system. Most of the swarfs and debris dropped into the collecting baskets at the ground.

Also during dismantling of the water separator, which was located on top of the core shroud, a small increase of activation in the exhaust duct of the suction device could be measured, but no other protection measures for the operational staff was necessary.

The most activated component of the reactor, the upper grid plate, had a dose rate of up to 50 Sv/h. It had to be evaluated, if the underwater plasma cutting technology would be appropriated to dismantle such a high-activated component. Five guide plates were cut off in 10 test and the radiological effects of the technique were analysed. Compared to components that were cut so far, the activation in the exhaust duct was distinctively higher, but again nearly no activity could be measured in the working area of the staff.

The amount and composition of the released radioactive aerosols and the efficiency of the installed suctioning device were investigated. The suction device over the water surface was tested again to work sufficiently. The achieved results were acceptable, so that the complete grid plate was dismantled with the plasma torch, assisted by a piercing saw (**Figure 26**).

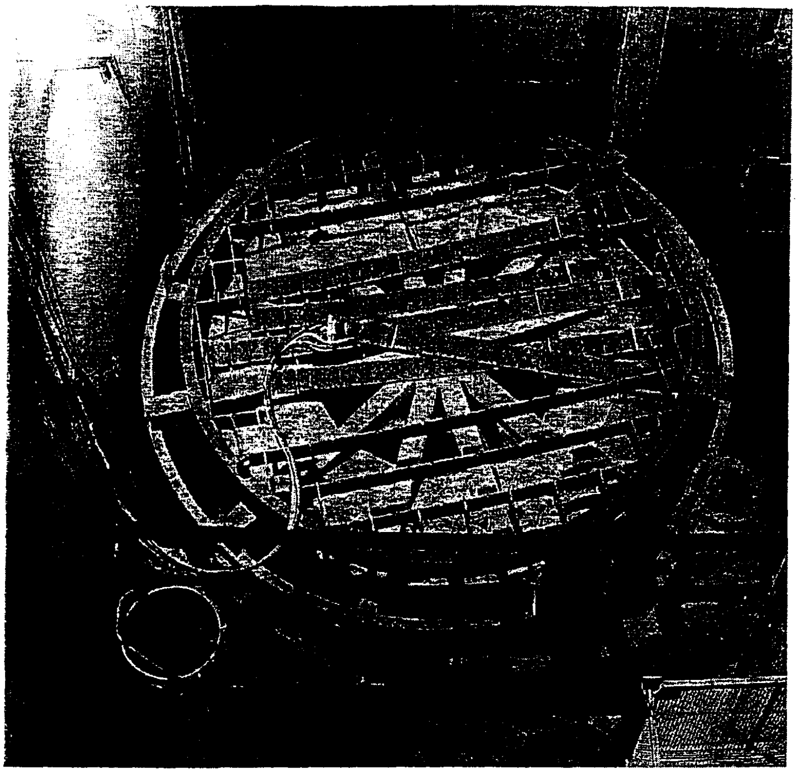


Figure 26: The upper grid plate was tested to be cut with the plasma arc technique

From this point of view it could be decided to dismantle the core shroud with plasma too, because the activity of this component is lower than that of the upper grid plate. The design of the dismantling unit could therefore be estimated for the use of tools with low restoring forces.

The function of the dismantling unit had to be tested and qualified in "cold tests" prior to the first use in the controlled area. It is planned to use this machine for the cleaning of the RPV walls as well. The manipulator is designed for being positioned on the core shroud as well as being clamped within the RPV. A high pressure water jet device can be mounted for cleaning.

The main intention of the qualification tests is to find hidden problems during handling and operation. Tests should therefore be as realistic as possible. The manipulator has been tested in a basin at the University of Hannover. The test program included basic function tests and operations needed for the core shroud dismantling. (**Figures 27–28**)

A mock-up of the core shroud was built to perform dry and wet tests. The positioning and handling of the machine was tested with a special test program, including simulation of the cutting movements, locking and release of the clamping device and the remote controlled change of the cutting tools. The plasma cutting of the windows for transportation of the ring segments was performed in underwater tests.

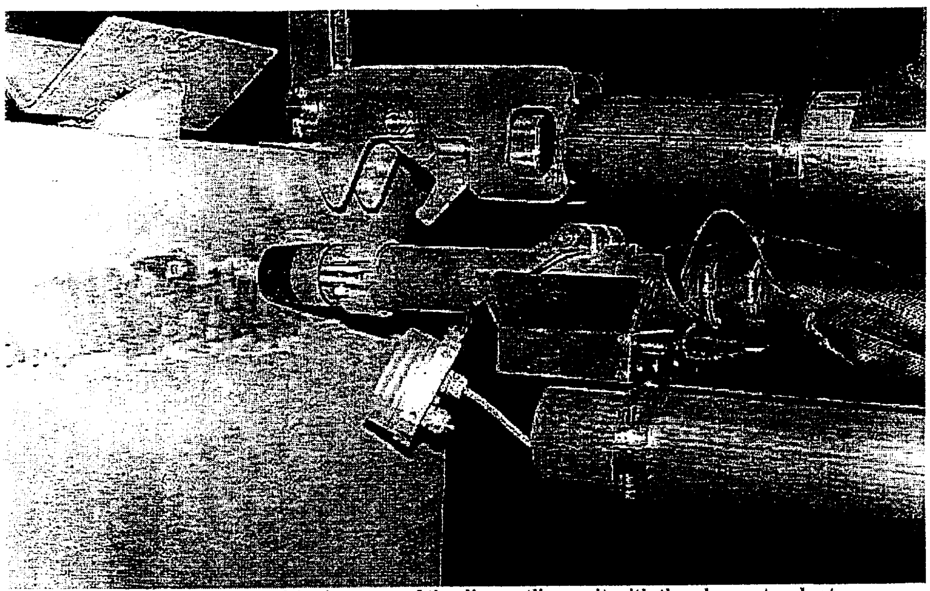


Figure 27: Positioning tests of the dismantling unit with the plasma torch at a mock up of the core shroud

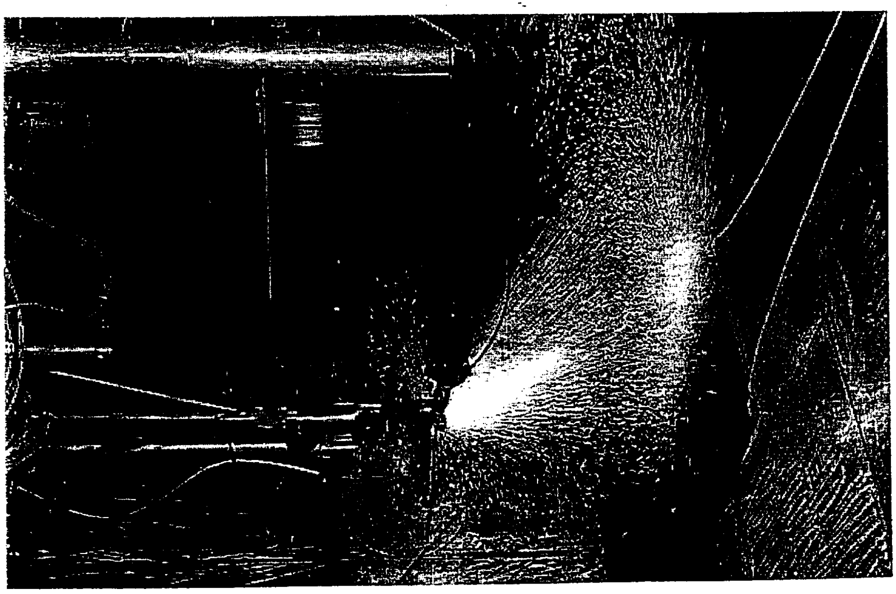


Figure 28: Cutting tests with the dismantling unit at the University of Hanover

Figure 29 shows the situation in the RPV. The dismantling device must be positioned on the cylindrical top of the core shroud in a water depth of 14 m at minimum. The last cut will be made in about 20 m of water depth. Once positioned on the core shroud, the cutting procedure can be observed by two under water cameras, which are mounted at the manipulator arm.

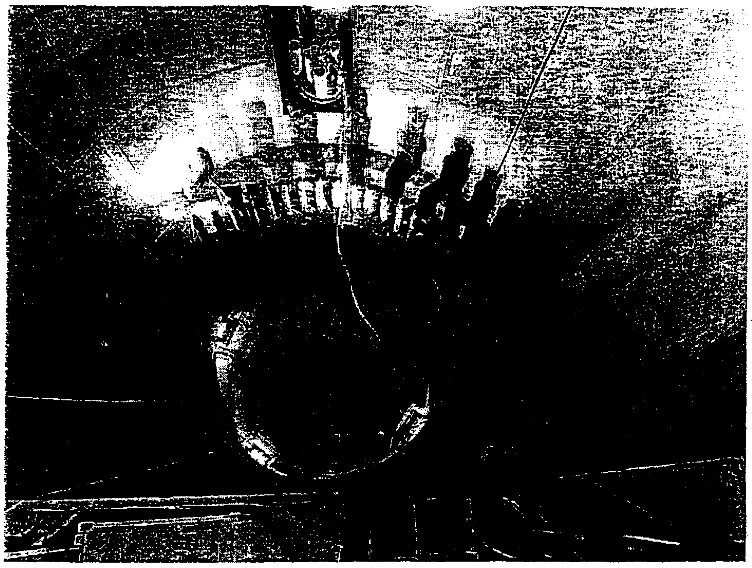


Figure 29: RPV after dismantling of the upper grid plate

B.3. Radiological survey and radioprotection optimization

B.3.1. Radiological characterization of RPV

BR3

As shown on **figure 30** underwater contact dose rates were measured inside the RPV. The high dose rate value at the bottom of the RPV is not due to high activity of the RPV material at that position, but to the presence, in the RPV, of secondary waste from the previous phase of the project:

- crud from the primary loop decontamination;
- chips from the in-situ thermal shield segmentation and from the internals dismantling.

This fact was confirmed after the post cutting cleaning of the RPV bottom.

In Belgium, 3 categories of radwaste are defined, namely:

- high level waste (HLW, contact dose rate > 0,2 Sv/h);
- medium level waste (MLW, 0,2 Sv/h> contact dose rate > 2 mSv/h);
- low level waste (LLW, contact dose rate < 2 mSv/h).

As the cost of radwaste conditioning and disposal increases drastically from category to category, it is essential to miminize the amount of radwaste of the higher categories. In practice, since the RPV will present dose rates (and activities) within the 3 different categories, it is important to fix as exactly as possible the category changing levels. Using samples of the RPV material extracted for other research purposes in 1995, it was possible to calculate the activity at the mid plane level and at the flange weld levels for the main material (CS) and for the SS cladding as well. The results of the activity measurement (only for ^{60}Co) are shown in **table 4**.

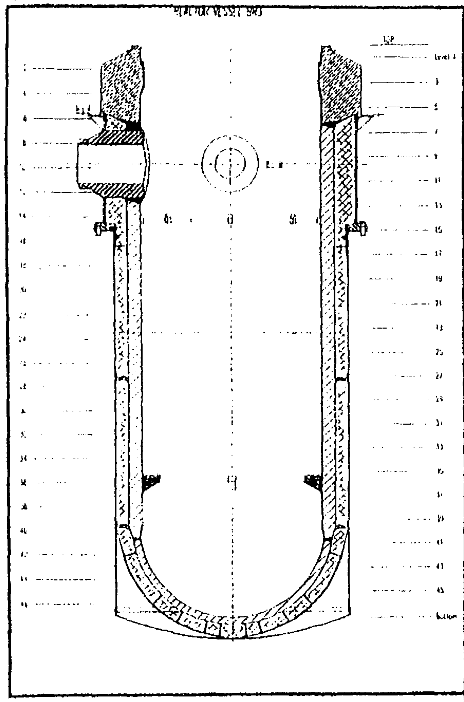
Table 4: RPV specific activity (^{60}Co)

Level	Mean specific activity of Base Metal (CS) Bq/g	Mean specific activity of the Cladding (SS) Bq/g	Factor SS/CS
Legs	7.78 E+02	9.71 E+03	12.48
Mid-plane	3.56 E+05	7.17 E+06	20.14

Both information were introduced to calculate the theoretical dose rate around the final package. This calculation led to the cutting plan showed on **figure 31**.

During the horizontal cutting, chips were collected in order to confirm the specific activity per level. During a cut the milling cutter produces chips in material at various material depths, so it is almost impossible to know where the chips are coming from. Therefore, this additional measurement only gave indicative value for waste segregation and characterization. Moreover, sometimes some chips can be coming from the internal cladding of the RPV which is made of stainless steel instead of carbon steel.

Contact Dose rates of the BR3 RPV under water (mSv/h)



Distance to the Mid plan (mm)	FFT Side	Hot leg Side	Cold leg 1 Side	Cold leg 2 Side	Mean
	Contact Doserate (mSv/h)				
2240	1,3				1,3
2040	1,3				1,3
1840	2,5				2,5
1640	3				3
1440	5				5
1240	15				15
1040	55				55
840	200	80	70	200	137,5
640	800	200	250	600	462,5
440	1500	800	800	1300	1100
240	2500	1800	1500	2500	2075
40	3500	2200	2200	2500	2600
-160	2500	2500	2500	2500	2500
-360	2000	2500	2500	2000	2250
-560	1000	1800	2500	1000	1575
-760	300	400	500	350	387,5
-960	70				70
-1160	35				35
-1360	9				9
-1560	8				8
-1760	15				15
-1960	70				70
-2160	400				400
-2360	800				800

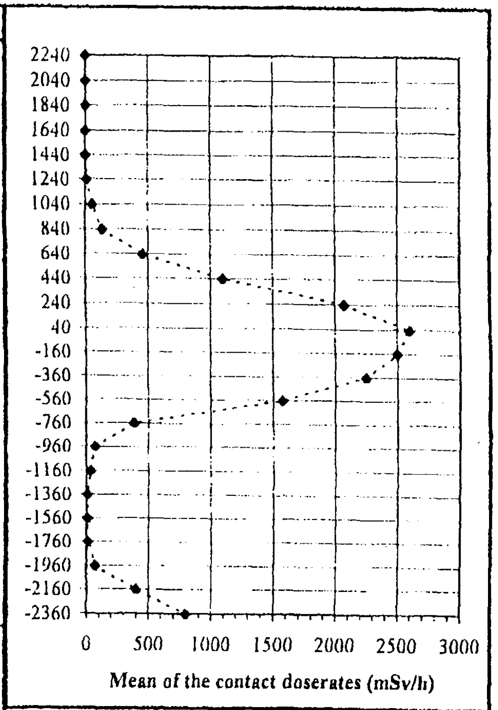


Figure 30

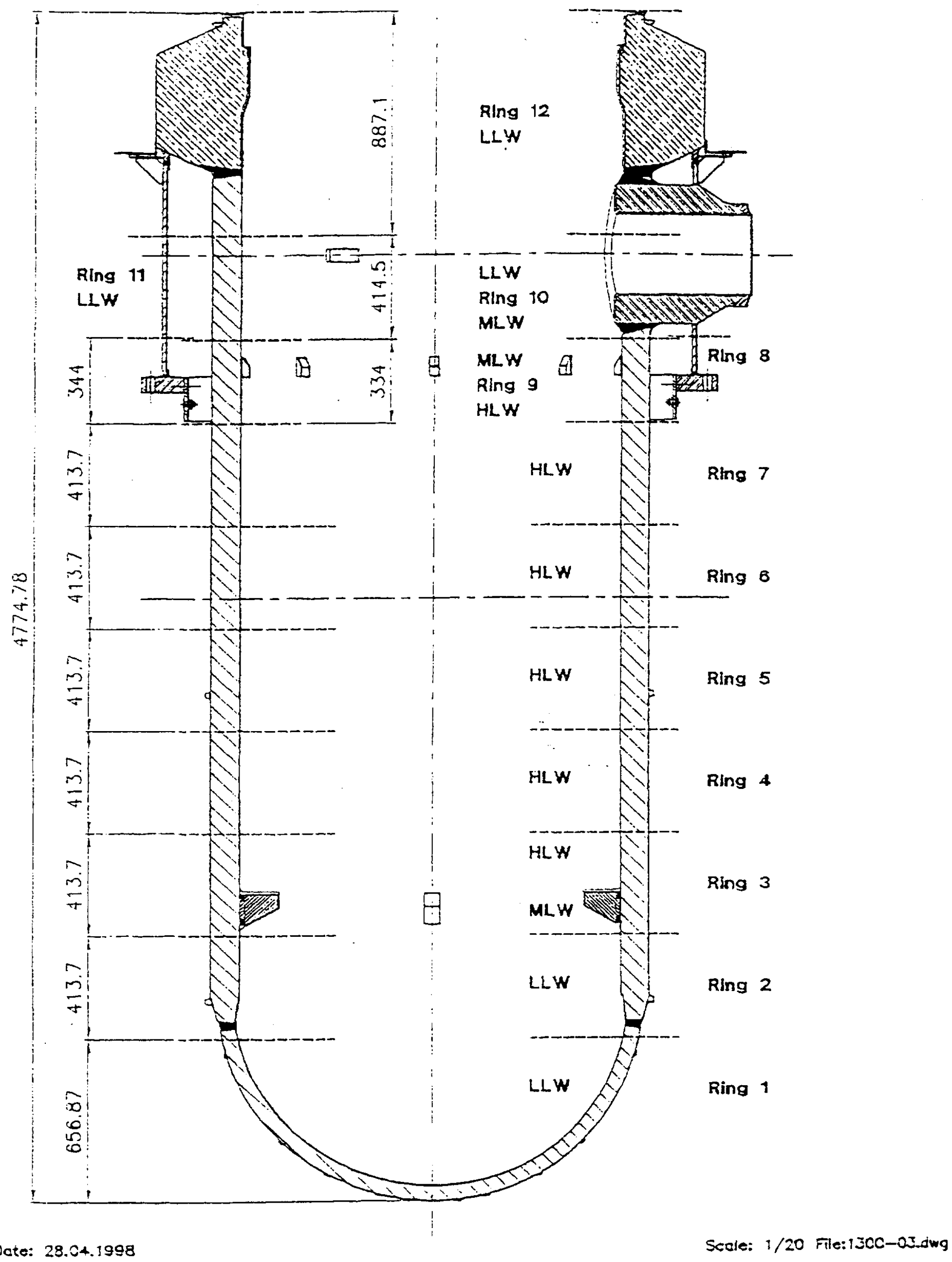


Figure 31: Different levels of the RPV horizontal cut

Table 5: Comparison of the difference between the expected and the actual waste categories of the cut rings

	Expected waste category	Actual waste category
Ring 1	LLW	LLW
Ring 2	LLW	LLW
Ring 3	MLW	MLW
Ring 4	**HLW**	**MLW**
Ring 5	HLW	HLW
Ring 6	HLW	HLW
Ring 7	HLW	HLW
Ring 8	**LLW**	**MLW**
Ring 9	MLW	MLW
Ring 10	MLW	MLW
Ring 11	LLW	LLW
Ring 12	LLW	LLW

KRB-A

The NIS company has performed a theoretical calculation of the activity for the whole RPV in the year 1983. Based on this study, **Table 6** shows the total activity inventory for some RPV sections calculated for the year 1997. For the selection of appropriated RPV cutting techniques the amount of aerosols which could be released at the cutting kerf is one important parameter. In general, the specific activity of the RPV wall in the core section is higher by a factor of 1000 than for the other RPV sections. In order to optimise the material flow, further sampling and activity measurements must be carried out to determine the exact separation area for such RPV material which can still be given for controlled recycling. It was found out that the activation in the upper and lower part of the RPV is < 200 Bq/g, which is the limit for controlled recycling by melting. At the core section of the RPV, the activation is about 3.1 E+12 Bq/g.

Table 6: Calculated Activity of the RPV for the year 1997

RPV section at level	Mass [tons]	Activity [Bq]	
		in the ferritic base material	in the austenitic cladding
flange	58.1	2.4 E+03	6.2 E+02
water separator	22.6	1.1 E+04	4.5 E+03
feedwater inlet	21.6	4.3 E+07	1.8 E+07
upper core	19.4	3.2 E+09	1.2 E+09
central core	34.5	3.1 E+12	1.3 E+12
lower core	22.0	3.7 E+09	8.4 E+08
recirculation inlet	39.8	5.8 E+09	1.1 E+09
bottom dome	42.0	-	-

Dose rate measurements were carried out at the outside of the RPV wall. The contact dose rate in the core section is about 63 mSv/h; in the upper and lower parts < 1 mSv/h.

To get a complete profile of the dose rate for the RPV, the program with radiological measurements near to the Biological Shield area has been started up. These values will be checked again after the removal of all RPV internals has been completed.

To determine the dose rate level at the working places and especially the influence of the different surrounding components, detailed measurements in the upper ring room (above the biological shield) have been carried out. The results are shown in **table 7**. The highest dose rates result from the primary steam line and the feedwater line. These tubes will be removed before starting the RPV-dismantling.
On the one hand, the RPV is still filled with water which reduces the dose rate, on the other hand RPV internals like upper grid plate and core shroud contribute to the increase of the dose rate outside the RPV. Due to this, the dose rate data will probably be verified after the removal of all RPV internals.

Table 7: Dose rate measurements inside the upper ring room 1998

	Contact dose rate [µSv/h]				
	0°	**90°**	**180°**	**270°**	**Average**
Primary steam line	1800	1380	1020	920	1280
Feedwater line	1400	880	910	810	1000
RPV wall	640	530	470	460	525
Shield cooler line	750	490	600	780	655

The upper grid plate was the highest activated internal component of the RPV. According to former calculations, the specific Co-60 activity was 7.7 E 08 Bq/g.
The specific activity of the RPV-plating in the core area is expected to be 1000 times lower. With respect to the results from cutting the upper grid plate with plasma, it should be possible to use thermal tools for the segmenting of the RPV too and to master the radiological protection measures during cutting.

Before dismantling the upper grid plate, two material samples have been taken with an underwater hacksaw for subsequent activity measurements (**see table 8**). The specific activity corresponds with the calculations. It is expected that the nuclide vector is similar to the one of the RPV cladding in the core section.

Table 8: Nuclide vector of two material samples from the upper grid plate in 1998

Nuclide	Cut 1 [Bq/g]	Cut 2 [Bq/g]
Co-60	3.08 +E8	1.27 +E8
Ni-63	3.22 +E8	1.28 +E8
Fe-55	6.7 +E7	2.4 +E7
Am-241	1.03 +E1	1.35 +E1
Pu-239 / 240	6.3 +E0	4.3 +E0
Pu-238	4.9 +E0	4.2 +E0

To check the radiological situation during the plasma arc cutting of high-activated components, 10 test cuts were performed at the upper grid plate with special regard for the hazard of incorporation.

The measurement program contained samples of the filter cake from the suction device upon the cutting area, aerosol measurement of the air in the working area, water samples and wiping tests at the plasma torch and the ODIN device. The results are given in **table 9**.

Table 9: Results of underwater plasma cutting at the upper grid plate 1998

	Co-60	Cs-137	Am-241
Aerosol measurement under the suction hood	500 - 800 Bq/m^3	< 1.5 Bq/m^3	< 0.2 Bq/m^3 (detection limit)
Aerosol measurement at the working area	3 - 8 Bq/m^3	0.2 Bq/m^3	< 0.1 Bq/m^3 (detection limit)
Water sample	290 Bq/l	40 Bq/l	9 Bq/l
Contamination of the cutting equipment	5,100 Bq/cm^2	17 Bq/cm^2	12 Bq/cm^2

The results showed that the aerosol concentration under the suction hood is high. However, the aerosol activity in the working area is neglectable. Anyway, an aerosol-surveying instrument was installed in the working area during the large-scale segmenting of the upper grid plate.

Using high pressure water jet has cleaned the primary steam pipes inside. This could reduce by a factor between 4 at the RPV-studs and by a factor of 2 inside the pipes the contact dose rate of the pipe.

An aluminium insulation foil surrounded the RPV. In 1998 material samples of the aluminium insulation were taken. The gamma activity was detected to be 310 Bq/g. In 1999 this insulation material was removed, including the position between the RPV and the biological shield. The material was compacted and packed into drums.

B.3.2. Radiological inventory of VVER reactor and internals

EWN

The activity of the reactor components and core components for the KGR was calculated for the work planning of dismantling. The neutron flux distribution was determined in radial, axial and vertical direction, proceeding from a model loading with fresh fuel elements.
The material of the core components are mainly low cobalt content, austenitic steels with considerable alloying addition of Ni, Mn and Cr as well as further alloying additions of C-steel with only low alloying addition (Fe content ca. 96 %).

Table 10: Mass specific activities for the reactor components of unit 1 (reference date 07/99)

Component	Maximum activity		Average activity	
	sum (Bq/g)	^{60}Co (Bq/g)	Sum (Bq/g)	^{60}Co (Bq/g)
CB core basket with facetted ring	2.1E+09	6.7E+08	2.0E+09	6.4E+08
CB core basket with cylinder	2.4E+09	7.5E+08	1.2E+09	3.7E+08
CB core basket with bottom plate	9.8E+08	3.0E+08	3.4E+08	1.0E+08
RPV core area with basic metal	3.0E+07	2.3E+06	1.0E+07	7.7E+05
RPV core area with plating (unit 3 and 4)	6.6E+07	2.2E+07	6.5E+07	2.1E+07
RC core area	5.5E+08	1.8E+08	2.5E+08	8.5E+07
RC cavity bottom with upper pipe unit	3.5E+06	1.2E+06	1.5E+06	5.0E+05
RC cavity bottom with pipe unit	3.7E+05	1.2E+05	7.8E+04	2.5E+04
PTU protecting tube unit with bottom plate	5.8E+08	1.9E+08	2.2E+08	7.0E+07
PTU protecting tube unit with pipe unit	1.7E+07	5.4E+06	2.7E+06	9.0E+05

Table 11: The nuclide vectors for austenitic and ferritic steels (reference date 07/99)

Nuclide	Unit 1	
	austenitic steel	Ferritic steel
^{54}Mn	<0.1%	0.6%
^{55}Fe	43.1%	87.2%
^{59}Ni	0.2%	<0.1%
^{63}Ni	23.1%	0.8%
^{60}Co	33.6%	11.4%

Proceeding from these activity concentrations and the corresponding nuclide vectors, the following maximum dose rates of the individual components can be calculated:

Table 12: Maximum dose rate (core centre)

Component	Dose rate (mSv/h)		
	50 cm	100 cm	200 cm
reactor cavity	2.9E-04	1.8E+04	8.4E+03
core basket	1.6E-05	9.7E+04	4.5E+04
protecting tube unit	2.6E-03	1.1E+03	4.1E+02
reactor pressure vessel	3.0E-02	1.9E+02	9.3E+01

B.3.3. Radioprotection optimization to cope with ALARA

BR3

➢ **ALARA principles**

The previous chapters described the technical part of the RPV dismantling. As far as radiological optimization is concerned, the ALARA approach has been implemented during the whole RPV dismantling operation.

The ALARA principle is the basic idea of the optimization of radiation protection and is based on 3 fundamental principles (ICRP - 26)

- No practice involving radiation exposure will be undertaken unless its use produces a net benefit (Justification).
- All exposures will be kept As Low As Reasonably Achievable with technological, economic, and social factors considered (Optimization).
- Exposures to individuals will not exceed the limits recommended for the appropriate circumstances (Minimization).

Thanks to our ALARA structure, that was set up at the beginning of the dismantling operations of the BR3 reactor, these three principles have been implemented in all stages of the decommissioning.

➢ **ALARA program elements**

The ALARA procedure consisted in the following steps:

- An initial detailed subdivision of the work procedure was made.
- 3 work areas were defined with their related average dose rate.
  - At the pool floor (to make some preparation work).
  - At the footbridge above the water level (to perform some manipulations with long handling tools)
  - At the operating deck (to do the maintenance and control the cutting tools).
- A prediction of the required manpower needed for each subtask in the procedure was made.

Using this information, a collective dose and workload prediction was calculated and discussed with the health physics department. It was also used to verify that all individual doses did not exceed legal limits.

This detailed analysis allowed the tasks giving the main contributions to the total dose to be identified and also the possibilities for reducing the radiation exposure at these critical points to be evaluated.

In order to analyse all data concerning dose and work duration, an electronic computerized dosimetric system is used. Using this data, the actual collective dose could be calculated for every subtask and compared with the predicted values as shown in **table 13**.

➢ ALARA practice for the RPV dismantling

The dismantling of the RPV, was optimized from a radioprotection point of view. Several options (levels of water, use of shielding material in general, remote controlled tools and long handling tools, decontamination need, decontamination way) were evaluated on the basis of a cost-benefit analysis for some of the operations to be performed.

It is obvious that working under water, as for the dismantling of the internals, decreases drastically the dose distributed to the workers (3 m water above highly active components provide a dose reduction factor of about 10^7) and, consequently, that most of the improvements must be done on maintenance operations (where the tools are removed out of the water), on the general waste management (disposal of the segmented pieces in the adequate zone, use of shielding, distance from the workers, ...), on the decontamination ability of the equipment and on the operations duration.

One of the important steps in such an operation is testing the equipment and training the operators in an unrestricted area in conditions similar to the ones they will face in the controlled area. This was impossible due to the fact that the principal cutting tools, like the bandsaw and the circular saw, already were become "hot" tools while used during the previous dismantling operations. Therefore the "cold" tests on a full size mock up of the RPV using these cutting tools had to be performed in the controlled area. The advantage of the RPV dismantling operation was that the operators were used to these tools and had already a lot of experience in working in such circumstances.

Today the RPV dismantling and cutting is finished. The remaining parts of the RPV like the hemispherical bottom, the cover and the insulation shield will be cut in the next dismantling phase using another cutting technique. In the following table a dose comparison of the different steps in the dismantling process of the RPV can be found.

The data in the fourth column *"First Estimate"* are predictions that were given in the global dismantling procedure (released in '98), covering all the different steps foreseen in the dismantling strategy of the RPV. The column *"Last Estimate"* gives the dose calculations as foreseen in the individual procedures (released just before the actual work has started in '99).

The following table gives an overview of the different steps in the dismantling process of the RPV.

Table 13: Differences between estimations and realized collective doses *(man-μSv)*

Operations	Actual man-μSv	Last Estimate man-μSv	Delta *(%)*	First Estimate man-μSv	Delta *(Estim. '98)*
Tests in controlled area					
First Series	2208	4330	*-49,0*	2397	*-7,9*
Second Series	2745	5000	*-45,1*	2310	*18,8*
Removal of pool sealing device	2456	1320	*86,1*	3317	*-26,0*
Desolidarization of the RPV					
NST side	7855	2500	*214,2*	6241	*25,9*
SOD side	2400	4680	*-48,7*	2370	*1,3*
Asbestos removal from core nozzle	3000	3285	*-8,7*	2614	*14,8*
Bolts removal of the RPV	1400	1956	*-28,4*	1564	*-10,5*
Restoring water tightness of the pool					
NST side	14658	4245	*245,3*	2019	*626,0*
SOD side	1400	1680	*-16,7*	2370	*-40,9*
RPV dismantling					
Insulation Removal	5022	8340	*-39,8*	6240	*-19,5*
Horizontal Cutting	4761	10080	*-52,8*	6780	*-29,8*
Vertical Cutting	3802	1188	*220,0*	6189	*-38,6*
Ring 8	443	3340	*-86,7*		
Ring 11	177	1032	*-82,8*		
Total	*52327*	*52976*	*-1,2*	*44411*	*17,8*

The doses shown in the column "last Estimate" are the doses like mentioned on the *"Volet A"*. This *"Volet A"* is a document to be annexed to all dismantling procedures. Every procedure concerning dismantling activities on activated and/or contaminated pieces is discussed with a responsible from the health physics department in order to optimize the collective doses. The Volet A has to be completed and approved before releasing the procedure.

The total collective dose differs slightly (- 1,2%) from the predicted value. Although when looking at the individual procedures, large differences can appear.

When the collective doses were overestimated, most of the time it was due to an overestimation of the workload. In most cases these tasks are more regular or recurring operations.

For the operations with an increased collective dose, the workload was higher mostly due to technical problems or unforeseen circumstances, as is explained hereafter.

➢ Tests in controlled area

These are the "cold" tests. They were performed in the reactor pool using the existing (contaminated) cutting tools. As these tests were done in the controlled area, a dose estimation had to be made prior to the execution. As a result of technical problems during the first series of tests, a second series of tests were necessary *(see § B.2.4 Full scale testing of techniques).*

➢ Desolidarization of the RPV - NST side

(ref. § B.4 Dismantling by various techniques)

The realized collective dose was more than 3 times the estimated dose in the procedure due to the following reasons:

- Preparation
  - During the preparation of the yard, some modifications were needed to the scaffolds (safety reasons).
  - To reduce the risk of α-contamination, an internal cleaning of the primary loops with water under high pressure was performed.
  - Due to a possible α-contamination distribution, a confinement had to be built around every loop to perform this cleaning operation.

  Initially, these three tasks were not foreseen in the procedure.
- Actual cutting

  The tool used to cut those RPV nozzles was specially designed for this task by an external company. Although it was tested during a long period in an uncontrolled area, it showed a lot of mechanical and electrical design failures that needed more preparation and maintenance (8 hours instead of 1 hour) for every cut than foreseen in the procedure. Therefore the actual cutting time was increased to 480 h instead of 151 h. The mean dose rate is the same (16 µSv/h).

➢ Restoring water tightness of the pool

The two main reasons for the difference in this operation are:

- Some discrepancies between the "as built" drawings and the reality made the position of these sealing pieces impossible at the time of the first attempt. Therefore, the design of the sealing devices had to be revised, and the sealing devices themselves had to be adapted; the placement of these devices had thus to be repeated twice with the corresponding dose uptake. The positioning of these devices was finally carried out in June 1999.
- Due to the modifications done on the sealing devices, part of the positioning operations was carried out at the NST level, i.e. close to the RPV in a higher radiation field.

> **Concerning the RPV dismantling:**

- Insulation removal:

  (ref. § B.2.3 System for the collection and filtration of swarfs and debris & B.4 Dismantling by various techniques)

  For the insulation removal, initially the dose calculation showed a collective dose of 4380 man-μSv. Some weeks after the start of this task, the procedure was reviewed due to several factors. A lot of extra operations were necessary, caused by the fact that the drawings were not 'as built'. A second factor was the insulation fibre and rust contamination in the water which needed also extra attention. Therefore new filtration systems were needed to solve the problem of the visibility of the water
  All these activities led to a new global dose prediction of 8340 man-μSv.

- Horizontal cutting:

  (see § B.4)

  For the horizontal cutting, the dose calculation was done using dose rates measured without any water in the pool. Therefore the dose rates were much higher than during the actual cutting work, where this was done under water. This resulted in a dose rate overestimation. In combination with a much faster cutting time (cutting speeds were 2 times faster than observed during the cold tests), the gain in collective dose was almost 53 %.

- Vertical cutting

  (see § B.4)

  For the vertical cutting the quite important increase in collective dose is due to a lot of unforeseen problems during the under water cutting. There were technical problems, problems with water pollution and visibility in the water. A lot of time is also lost during the manipulation of the cut pieces to put them in the transportation racks.

- Ring 8
  During the cutting of Ring 8, which is the support skirt ring of the RPV, it appeared that this piece was not low active waste like it was characterized before, but medium active waste. This had an influence on the final dimensions of this piece. To minimize the waste quantity, it was indispensable to produce smaller pieces. This resulted in some technical problems and therefore it was necessary to review the procedure. The best option (radioprotection point of view) was by modifying the fixation system on the turntable and cutting Ring 8 under water like the rest of the RPV. However this would be a very time consuming work with an extreme delay in the planning and an extra dose uptake. Therefore the health physics department agreed to perform the 8 extra cuts in the dry cutting workshop using a bandsaw. The contact dose rate was less than 3 mSv/h. The operators were only close to the piece during the clamping on the bandsaw. This also can

be seen as an ALARA approach, as the dose is kept as reasonably low as possible taking into account the economical factor.

- Ring 11

  The cutting of Ring 11 is also done in the cutting workshop using a plasma arc torch. This was already foreseen in the actual procedure. It was acceptable due to the rather low dose rate in contact (>2 mSv/h) and it could be cut rather fast. The decrease in collective dose (> 80 % for both operations) can be attributed to an overestimation in workload. Like mentioned before, this is often the case for such tasks with experienced operators.

➢ **Conclusion**

Although dismantling tasks are quite specific, compared to maintenance and operational tasks, the ALARA programme has been proved to be totally enforceable if such an operation is prepared, followed and analysed from a radiation protection point of view The ALARA principle has been incorporated into all levels of decision-making which occur during the RPV dismantling operations.

<u>EWN</u>

The planning of radiation protection for the dismantling of unit 5 is based on the following dose commitment limits:

- 1 mSv/day
- 10 mSv/year.

For areas in the steam generator room, the following dose rate limits are valid:

- 300 μSv/h for areas of intervention
- 100 μSv/h for accessible areas.

The complete dismantling excluding maintenance is performed remotely from a control panel. At this control panel, all important actual dose rates are shown.

The estimation of radiation exposure for the dismantling personnel in the steam generator room takes account of the following conditions:

- dose limits
- equipment used
- local and air-technical separated realisation of cutting activities (dry cutting station and wet cutting station)
- shielding and air-technical zones in dependence of contamination and dose rates use of the present building structure for shielding

- limited exposure time (one shift working by 6 h presence with an average of 21.5 working days per month).

The estimation of the time for dismantling of the reactor components inclusive the assembling/dismounting of the cutting equipment leads to a duration of ca. 30 months for two reactors. As a result of this, there is a collective dose of 1.8 Sv for two reactors.

Aerosol production

During the cutting of the reactor components, aerosols will be produced. The authority has defined the following limit values for the release of aerosols:

- 5E+10Bq/year (long-term release)
- 5E+08Bq/day (short-term release).

For reactor cutting, only 30 % of the release values should be reached. The aerosol emission is mainly dependent on the used cutting procedure as shown in the table below.

Table 14: Aerosol emission rates of selected cutting procedures

Cutting procedure	Dry cutting	Wet cutting
band saws	3 - 22 mg/cm²	0.003 - 0.2 mg/cm²
CAMC-arc cutting	-	1.2 -2.9 mg/cm²
plasma cutting	70 - 220 mg/cm²	0.7 -2.2 mg/cm²
oxyacetylene cutting	70 - 220 mg/cm²	-

Based on a ratio of 9:1 for the mechanical and thermal cutting times in the cutting places of the steam generator room, the following specific aerosol emission rates for the main part of reactor components are obtained:

- dry cutting 42 mg/cm²
- wet cutting 0.47 mg/cm².

Considering also the following factors:

- average mass specific activity of the cut material;
- dimension of cutting surface of the activated part of the reactor component;
- cleanup efficiency of exhaust air filtering of 99.9978 %.

The following aerosol activity results from the individual reactor components.

Table 15: Aerosol activity results from the individual reactor components

Reactor component	Aerosol activity (Bq)
core basket	2.4+E06
reactor cavity and cavity bottom	3.7+E06
protecting tube unit	9.7+E05
reactor pressure vessel	5.1+E06

The total release of 1.2E7 Bq remains by magnitudes under the licensed annual release values for aerosols.

Maximum short-term release of aerosols can be expected in the dry cutting station during the cutting of an RPV-core segment. On the assumption that 3 segments will be cut vertically in the post-cutting place of the dry cutting station within a working day, an aerosol release of 5.3E4 Bq/day will result, which is below the licensed short-term release value.

KRB-A

Radiation exposure for the dismantling stuff must be as low as possible. A reliable way to reach this aim is to cut the components under water by thermal or mechanical techniques. This method is especially suitable for segmenting the reactor internals in the most cases. Cutting under water in combination with a suction device (**Figure 32**) upon the cutting area reduces the emission of aerosol to a minimum and so the aerosol activity in the working area is negligible.

The experience gained from cutting of the reactor head showed that the segmenting in air with thermal tools is possible, if an effective suction and filtering device is used.

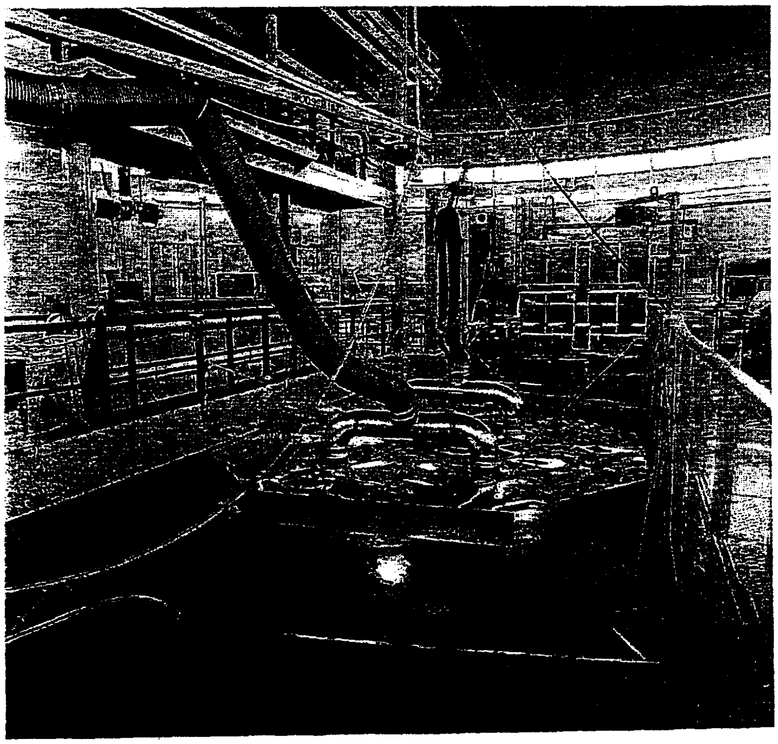


Figure 32: Suction device for thermal cutting under water

Another possibility to reduce the radiation exposure is given by remote controlled or automatically working tools. Also handhold tools can be efficient, because they often needs no time for set-up and help to minimize time of exposure.

After draining water from contaminated components, problems have to be taken into account referring to the normal drying procedure at the atmosphere, which will cause a substantial release of aerosols into the working area. Therefore the inner surface of the RPV will be cleaned before cutting with a high pressure water jet to remove the contamination layer as far as necessary to clean the areas around the vessel above and below the bioshield from a wide-spread dust layer. This substantially reduced the hazard of aerosol incorporation for the dismantling staff.

Due to the high contamination of the tools and supporting devices, they have to be cleaned carefully when lifted out of the water. This is necessary, because of the radiological signification of the alpha radiator in case of incorporation.

Radiological measurements at the RPV have been executed by making use of a Tele-Gamma-Scanner at different areas at the RPV outside. A measuring campaign with such an automatic and remote-controlled Tele-Gamma-Scanner has been performed by the NIS company for localisation and evaluation of hot spots in the radiation field of the RPV. The main advantage of the Tele-Gamma-Scanning System is the possibility to detect certain radiation areas automatically by positioning the equipment far away from the radiation source. This will lead to a dose saving for the measuring campaign.

B.4. Dismantling by various techniques

BR3

➢ **Preliminary operations**

These operations were executed with a dry refuelling pool, the RPV still being in its cavity, under the bottom of the refuelling pool. So the access to the pool floor was possible but had to be reduced as much as possible for radioprotection reasons. **Figure 33** shows a close-up of the RPV connection to the BR3 installation.

Desolidarization of the RPV from the bottom of the reactor pool (see figure 33)
The selected process for cutting the bottom of the reactor pool is the plasma arc torch handled by an operator. The cutting has to be done quickly for limiting the dose uptake of the operations (radioprotection optimization). In addition to this desolidarization, some cuts at the bottom of the reactor pool were also needed to give access to the fastening bolts of the RPV support flange, to give access to the hot and cold legs thermal insulation and to allow the installation of the sealing equipment for the future water tightness of the pool.

Removal of the asbestos situated around the primary pipes near the RPV (figure 33)
SCK•CEN personnel carried out this operation, as the nuclear hazard was estimated to be far above the asbestos hazard. Nevertheless, to avoid the spread of asbestos fibres, a double confinement was installed in the RPV pool (**figure 34**).

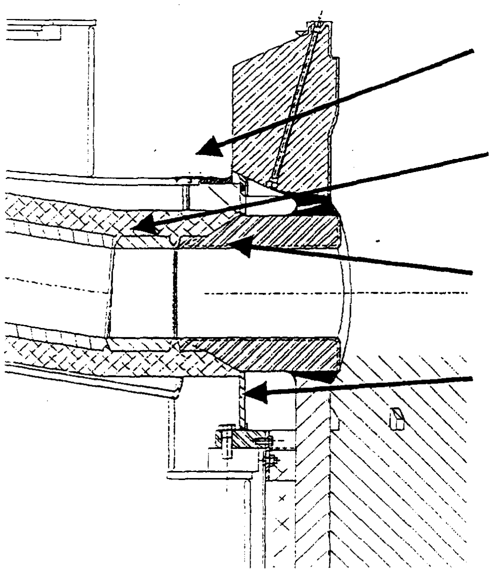


Figure 33:

Desolidarization of the RPV from the bottom of the reactor pool

Removal of the asbestos situated around the primary pipes near the RPV

Desolidarization of the RPV from the hot and the cold legs (cutting the primary pipes near the RPV)

Unbolting from the NST (its supporting structure)

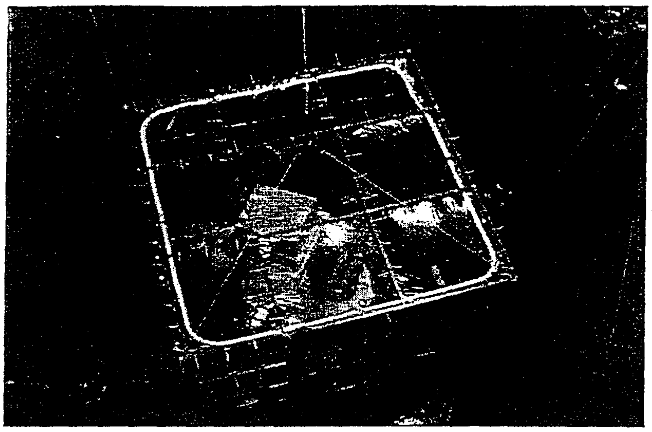


Figure 34: First of the double confinement installed in the RPV pool

Desolidarization of the RPV from the hot and the cold legs

- Cutting of the primary pipes at the outside of the bioshield (**figure 35**).

  The main operation is the cutting of the pipes at the RPV flange level. Regarding the very tight space available to perform this operation, access was needed through the primary pipes at the bioshield side. This operation was carried out with a quite common automatic pipe cutter, using two lathe tools diametrically opposed.

Figure 35: Cutting of the primary pipes at the outside of the bioshield

- Cutting the primary pipes near the RPV (**figure 33**)
  This operation was delicate due to the fact that access was only available at the inside of the piping. We thus developed, with an industrial partner, an automatic milling cutter able to cut the necessary thickness. The challenge was to have a machine fitting into a diameter of 254 mm, able to cup up to 110 mm wall thickness. Finally, it was decided to make a second cut of the primary pipe connections just above the support flange of the RPV in order to get access to all the RPV fastening bolts. The cutting tool is an automatic milling cutter with a diameter of 30 mm for the first part of the cut, 25 mm for the second, deepest part.

The intervention procedure was based on the results of the cold test with the internal pipe cutting tool (see chapter B.2.4). The results can be summarized with the following data.

- Cutting diagram: the milling cutter went through the piping wall following a spiral shape with a cutting depth of 15 ~ 20 mm, according to the diameter of the tool.
- The quantity of cutting tools required for the intervention was a bit greater than foreseen:

Table 16: Results of the cold test with the internal pipe cutter *(see also §B.2.4)*

Cut nr	Qty used	Qty foreseen	Item to be cut	Cutter (Øe x length) [mm]
1	1	1	Hot leg, thickness ~ 50 mm	25 x 90
2	1	1	Hot leg, thickness 119 mm	25 x 90
	1	1		20 x 125
3	3	2	Cold leg, thickness ~ 40 mm	25 x 90
4	2	2	Cold leg, thickness 108 mm	25 x 90
	3	2		20 x 115

- The rotating speed was taken equal to 300 ~ 380 rpm, according to the diameter of the tool; a reduction of the speed was required while cutting in stainless steel.
- The tangential feed at the tip of the cutter had to be maintained at a constant value of about 35 mm/min.
- The radial feed was such as to keep the cutting depth of the tool at a constant value while following the required spiral shape.

Initially the duration of the intervention was estimated at 10 days/1 team; in fact the duration was twice as much with 10 days/2 teams. This was mainly caused by the weaknesses of the equipment built to fit into the very small diameter of the primary piping system ($Ø_i$ = 250.8 mm for the cold legs) and which had to cut a pipe thickness of more than 100 mm.

As an example after each cut, a control and maintenance of the tool was planned: the foreseen duration of 1 hour for that point was often trespassed (up to 4 to 8 hours). The environment was highly hostile due to alpha contamination in the piping: working close to the tool required always the wearing of overalls and gas masks as a minimum (**figure 36**).

Figure 36: Cutting tool during maintenance (lower level of the mobile footbridge)

Desolidarization of the RPV from the NST (figure 33)

The selected procedure to remove the 24 fastening bolts of the RPV on the Neutron Shield Tank was the pneumatic unbolting. Due to the high level of corrosion at the bolts level, this operation took about three times more than foreseen (and thus increased the total dose uptake!).

Reinstallation of the water tightness of the NST and the reactor pool

As the RPV and its primary pipes were part of the pool leak tightness system, we had to seal the openings made by the primary pipe cutting, within a very tight space. The operation was carried out with an industrial partner, who developed a system based on an epoxy-based polymer and form-shaped sealing system. Cold tests were carried out on a real scale mock-up and everything was ready for the installation at the end of March 1999. Unfortunately, some discrepancies between the "as built" drawings and the reality made the positioning of these sealing pieces impossible. Therefore, the design of the sealing devices had to be revised, and the sealing devices themselves had to be adapted. The positioning of these devices was finally successfully carried out in June 1999 (see **figure** 37).

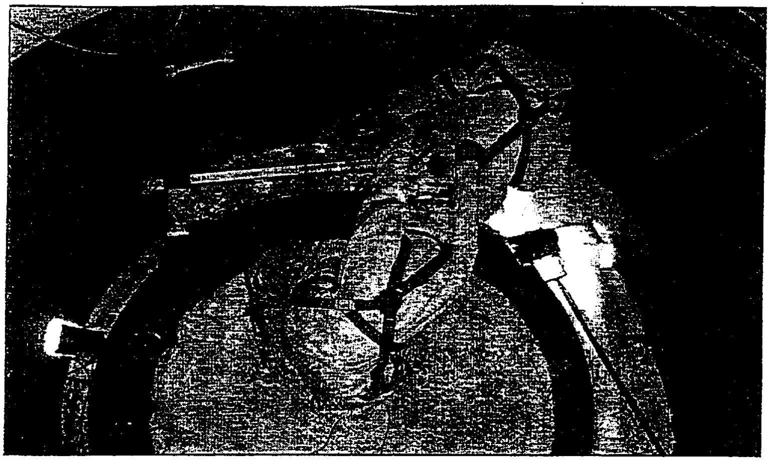


Figure 37: Positioning of sealing devices

Finally the RPV was ready to be lifted. A guiding system was also installed, as the mechanical clearance between the RPV and the sealing devices was less than 10 mm. The pressure vessel (28 tons) was then lifted up in one day, using a new gantry crane installed above the RPV pool (see chapter B.2.2. and figure 12). The water level in the pool was raised at the same pace as the RPV lifting.
On August 24, 1999 the RPV was removed from its position into the reactor pool for further dismantling.
Direct broadcast was assured thanks to a webcam. The operation is still visible at the address:
http://www.sckcen.be/eccdecommissioning/europe/pilot/br3/dismant/vessel/anim.html

Removal of the insulation shell
The insulation shell was bolted to the RPV through two profiles and on the upper side, it was bolted to the RPV supporting skirt. It was necessary to remove 60 bolts to free the insulation shell from the RPV. Because of the horizontal position of these bolts, they had to be drilled by a remote hydraulic cutter. For reaching easily the different levels at which the bolts were placed, the remote hydraulic cutter could move up and down along a beam. Here again, cold tests were carried out.

For the actual work, it became almost impossible to localise the screw heads due to a high level of corrosion on the shroud surface. It was then decided to cut the entire circumference of the core shroud using the hole cutter machine. Therefore, it was finally necessary to drill 10 times more holes than foreseen. An additional problem occurred: some turbidity of the water appeared. This was due to rust but also to the thermal insulation which became breakable into something like dust. Sometimes, the visibility was so bad that the operation had to be stopped. This problem was solved putting into service additional filtration and purification facilities.

Removal of the insulation and the fastening profiles of the insulation shell
The insulation shell was bolted on the RPV by T-shaped fastening profiles and connection pieces on two levels. Between and on top of these fastening profiles, there was the thermal glass fibre insulation, fastened with a metal mesh. The insulation was also held together with metal strips. On the bottom side of the RPV the insulation was tightened to the RPV with eight strips. The strips were attached on the RPV by bolts throughout the insulation material.

As the mesh was totally rusty, the removal of the insulation was done using a long handling tool. The so liberated insulation fell into a fishing net previously installed on the floor of the pool. By remotely closing the fishing net, the insulation was taken out of the water and evacuated as standard low level waste.

In a first study, it was foreseen to unscrew the bolts of the fastening profiles of the insulation. Finally, the T-shape profiles were remotely attached to the plant container crane and torn up. As the fastening profiles were low activated, their further dismantling was done by hand held tools.

- **Turbidity of the pool water**

Visibility
During the removal of the reactor shroud and the thermal insulation, a strong pollution of the reactor pool occurred. The high water turbidity refrained the continuation of the dismantling works. Actions were taken to analyse the origin of the pollution, its nature and to define the corrective actions.

Analysis of the pollution
During the dismantling period from October to December, we observed an increase of the water conductivity from 4 µS/cm (demineralized water) up to 63 µS/cm, the pH increased also from a slightly acidic value of 5.6 up to 8.5. In the periods of bad visibility, the water was slightly coloured yellow and particles in suspension were measured at a concentration of about 4 mg/l. The activity of the water increased slightly but remained lower than 0.5 Bq/ml in ^{137}Cs and ^{60}Co.
The evolution of the water pollution is shown on **figure 38** which gives the water conductivity and the pH observed during the whole period.

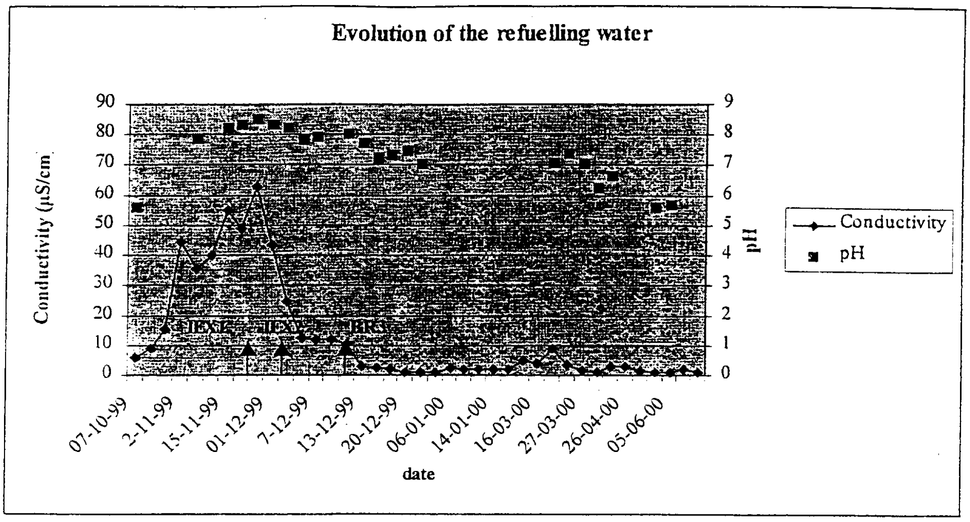


Figure 38: Water conductivity and pH observed during the whole period

During the shroud drilling, the main elements found in solution were B (54%), Si (29%) and Na (5%). The suspended matter was essentially Fe (90%) and Zn (5%). At that time, no significant quantity of insulation fibres was found in suspension.
During the removal of the insulation material, the conductivity was at its highest value (between 45 and 60). The main elements found in solution were B (55-80%), Si (8.5-30%), Na (1-10%), Ca (1.9-2.3%) and K (0.3-1%). The suspended matter was essentially Fe (64-95%), Ca (3-10%), Zn (3-4.7%), Si (3-6%), K (1-2%).
The origin of the pollution was at the beginning clearly the release of corrosion products in the water during the shroud drilling: the iron hydroxides rust from the drilling of the carbon steel shroud and the zinc hydroxide from the galvanized wire net used for the fixation of the insulation. Afterwards, when the insulation material was removed, mineral fibres were found in the water; some binding products, mortar like, used for the fixation of the insulation were also probably released in the water. No organic pollution or bacterial growth was observed significantly.

Cleaning actions
To recover clear and pure water, it is necessary to remove the particles in suspension and the dissolved ions. Laboratory tests showed that mixed bed ion-exchangers were able to remove the dissolved ions and recover a low conductivity. Filtration tests were performed at pilot scale using 10" filter cartridges of different porosity (10, 1.2 and 0.65 µm). The results showed that it was necessary to use at least a 1.2 µm filter.

The existing purification system of the water of the refuelling pond comprises a filter unit of a capacity of about 20 m^3/h and a 210 l ion-exchange column of a capacity of 2.5 m^3/h.

The filters used are 10" wound 1μm polypropylene filter cartridges (64 filters). The ion exchanger is a homogeneous mixture of a strongly acidic cation resin and a strongly basic anion resin.

This system is insufficient to deal with the strong pollution observed; moreover it appeared that the ion exchange column was saturated. Therefore, several actions were undertaken:

- installation of an additional mobile filtration unit of a capacity of 20 m^3/h pumping directly in the pool (IEX1 on figure 38);
- replacement of the saturated resins (IEX2 on figure 38);
- installation at the outlet of the mobile filtration unit of two mobile ion exchange columns of a capacity of 2 to 3 m^3/h (BR3 on figure 38).

All these systems were progressively put into operation so that the situation could finally be controlled around mid-December. Afterwards, the visibility could be kept under control except during short periods of peak pollution corresponding to release or resuspension of rust or fibres. To maintain a high water quality, it was necessary to replace regularly the saturated filters or the resins. During the whole cutting period of the reactor pressure vessel, the production of secondary waste amounted to about 2.9 m^3 of burnable cartridges (25 charges) and 0.7 m^3 of burnable ion-exchangers.

➢ **Horizontal cutting of the reactor pressure vessel – Circular saw (figure 39)**

On the basis of the tests (see chapter B.2.4.) some improvements were made:

- the use of sawblades with a much smaller eccentricity allowed to increase the feed speed as well as the life span by a factor up to 4;
- a semi-automatic system for the recuperation and evacuation of the metal chips avoided a complete jam of the circular saw during operation; moreover it was necessary to place a cyclone before the inlet of the suction pump as a protection against a flow of too much chips at a time.

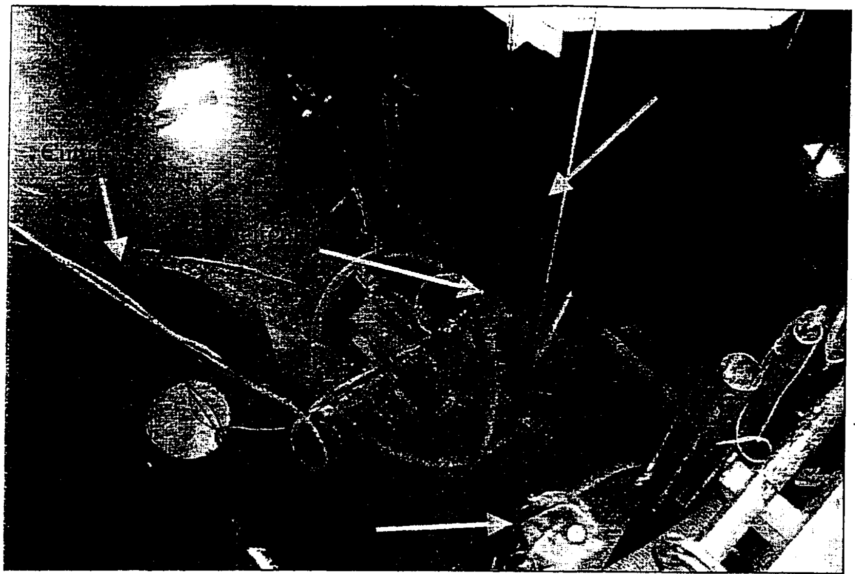


Figure 39: Sight of the reactor pool with the RPV, the saw on its XY table, the metal chips evacuation system with the cyclone and the already cut rings on their storage structure

The results of the intervention can be summarised as follows (see also table 17):

- Cutting diagram: a complete horizontal cut of the vessel was achieved in two steps:
  - linear cutting of introduction sectors with the translation of the circular saw working on a fixed vessel;
  - circumferential cutting of the remaining material with rotation of the vessel with a fixed circular saw.
- As the life span of the sawblades improved significantly, the quantity of cutting tools required for the intervention was much smaller than foreseen: 9 blades made according to DIN 1838C with an external diameter of 500 mm and a tooth pitch of 19.63 mm.
- The rotational speed of the sawblade was maintained constant around 10 pm.

The foreseen duration of the intervention was estimated at 107 shifts on the basis of a mean feed speed of 15 mm/min; in fact the actual duration of the intervention was of 65 shifts leading to a reduction of 42 shifts, mainly because of much better performance of the sawblades (feed speed up to 80 mm/min and longer life span).

This duration does not take into account the time required for the following operations:

- the maintenance of the turntable prior to the intervention (3 shifts, 116 man-hours);
- the handling of the hemispherical bottom head of the reactor pressure vessel (3 shifts, 80 man-hours) to its temporary storage;
- the development of a supplementary cyclone water filtration system to protect the suction pump against the swarfs: the built-in cyclone of the pump was not able to follow the production of swarfs and became obstructed more than once (5.5 shifts, 161 man-hours);

- the vertical cutting of ring RPV-8 into pieces with the band saw (3 shifts, 78 man-hours), see the chapter on vertical cutting, hereunder.

Table 17: Reactor pressure vessel dismantling – horizontal cuts

Steps	Predicted			Actual			Cut surface	Av. Cutting speed	Cutting capacity	Swarfs	Number of blades
	Shifts	Workload [H x h]	Dose [H x mSv]	Shifts	Workload [H x h[	Dose [H x mSv]	[dm²]	[mm/min]	[mm²/min]	[kg]	
Preparation	3	72	0.444	7.0	159	0.432			-	-	
Ring 1	11.5	372	1.788	14.0	579	0.889	31.5	22.0	357.6	15.1	2
Ring 2	13.5	324	1.108	8.0	254	0.432	56.8	28.5	701.9	27.3	1
Ring 3	12.5	300	1.028	4.5	116	0.293	56.8	36.7	890.0	27.3	1
Ring 4	11.5	276	0.876	4.5	128	0.201	56.8	31.7	810.0	27.3	1
Ring 5	12.5	300	1.028	7.5	199	0.430	56.8	25.5	711.6	27.3	1
Ring 6	12	288	0.952	5.5	144	0.253	56.8	28.7	891.4	27.3	1
Ring 7	12.5	300	1.028	5.5	137	0.258	56.8	27.4	543.9	27.3	1
Ring 8	5	120	0.54	2.5	59	0.079	12.4	41.2	791.7	5.9	(1)
Ring 9	12.5	300	0.992	6.0	134	0.385	56.8	21.0	-	27.3	1
Total	106.5	2652	9.784	65.0	1908	3.652	441.3			211.8	9

➢ Vertical cutting of the RPV – Band saw

This paragraph considers only the rings that were cut with the bandsaw. These rings are showed in **figure 31**.

Ring 1 (bottom of the vessel) and also the vessel head will be cut with another technique later on in the dismantling project. Ring 11 (low activated) was cut with the plasma torch in our dry cutting workshop and ring 8 (also foreseen as low activated) with a reciprocating saw (hands on techniques).

- *Cutting of the vessel flange*

After the last horizontal cut, the first piece presented to the bandsaw was the reactor flange (ring 12 and 10; see **figure 31**).

First 15 vertical cuts of about 950 mm length were performed in the vessel flange in order to make 'teeth'. The last cut was done above the hot leg in order to turn the sawblade from a vertical cutting position into a horizontal cutting position at a well-defined height. Then a hydraulic motor drove the turntable so that the bandsaw could make a horizontal cut and cut off the 'teeth', one by one. The cut pieces were put immediately in standard 400 l-drums and could then be removed as low activated waste (**see figure 40**).

The remaining ring (ring 10, **figure 31**) could then be cut in the same way like the other rings.

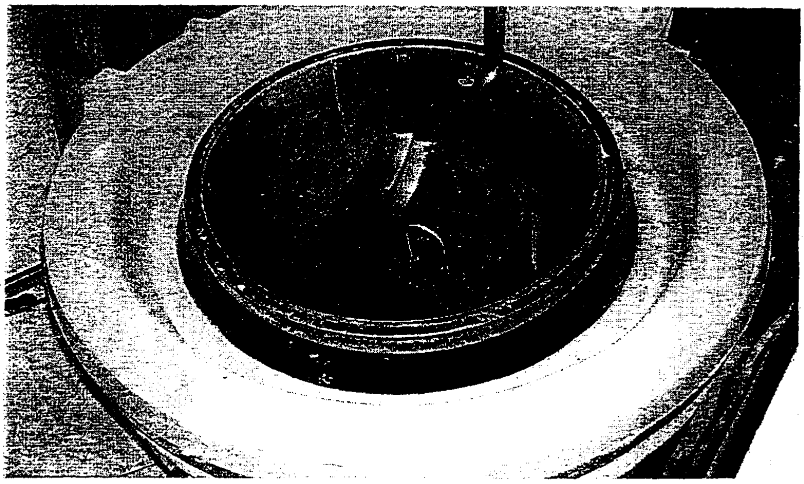


Figure 40: Standard 400 liter drum loaded with a flange piece

The vertical 'teeth' cuts were quite difficult for two reasons. First due to the thickness of the flange (up to 360 mm) and second due to the presence of a big weld and the sometimes complex cross section. This is reflected in the high consuming rate of the blades (17 against 3 for a normal ring – see **table 18**).

- *Cutting of the rings*

The cutting of the remaining cylindrical rings of the RPV shell showed no problem. However, the used cutting speed was much lower than during the cold testing (9.5 mm/min instead of 19 mm/min. This can be explained as follows: for the two first rings, the operating team used a lot of sawblades. In order to avoid the same high consumption of blades during the next rings, the operating team started the next cuts with a reduced feed compared to the feed used during the cold tests. This went well so that the team would not take any risks and remained at the same cutting speed.

- *Summary*

The table on the next page gives the most important data concerning the ***vertical*** cutting

One remarks the great underestimation of the workload and the collective dose.
The causes were:

1) The unforeseen visibility problems due to the presence of the thermal insulation (the same problem already encountered during the horizontal cutting).
2) The lower cutting speed that was used (50% lower) and the fact that the blade had to be changed more often than foreseen, doubled the cutting time for the ring.
3) But the main cause was a totally wrong estimation of the waste manipulation after the cutting of the ring, i.e. the filling of the waste racks, and certainly when there were problems with the weak designed guiding pins (see further: encountered problems).

Table 18: Data concerning vertical cutting

Ring	Predicted			Actual			Av. Cutting speed (mm/min)	Cut surface (dm^2)	Cutting capacity (cm^2/min)	Swarfs (kg)	Number of blades
	Shifts	Workload (man-hr)	Dose (man-mSv)	Shifts	Workload (man-hr)	Dose (man-mSv)					
Preparation	-	-	-	3	55	0.124			-		
12 (vert)	9.5	163.25	0.368	32.5	669	1.219	4.7	424.0	29.61	66.14	17
12 (hor)							22	57.8	10.42	9.13	
10	2.5	43.5	0.127	13	254	0.435	10.4	55.8	11.9	8.82	6
9	2.5	42.75	0.124	6	147	0.207	11.0	53.1	13.0	8.39	3
7	2	34.8	0.095	9	196	0.345	8.0	56.2	9.0	8.89	3
6	2	34.8	0.095	7	160	0.356	9.4	55.8	10.7	8.82	3
5	2	34.8	0.095	4.25	65	0.135	9.0	55.8	10.3	8.82	3
4	2	34.8	0.095	5.25	116	0.360	7.8	55.8	8.9	8.82	3
3	2	34.8	0.095	5.5	75	0.222	10.6	55.8	12.1	8.82	2
2	2	34.8	0.095	5.5	134	0.404	8.9	41.4	10.2	7.85	3
Finishing	-	-	-	1.5	33	0.049			-		
Total	**26.5**	**458.3**	**1.189**	**96.0**	**1905**	**3.856**		**912**		**145**	**43**

Remarks
- With the average cutting speed is meant the average speed of the saw blade when it goes through the base metal.
- With "preparation" and "finishing" is meant the start and termination of the cutting yard.
- Only the rings cut with the bandsaw were taken into account.

➢ Encountered problems during the cutting of the RPV

Although the bandsaw technique proved its reliability during previous dismantling phases and the cold tests did not came up with any major problem, we encountered few problems from which some lessons had to be learnt.

- Visibility problem

  After the removal of the insulation shell and the thermal insulation, a part of the thermal insulation remains between ring 10 and ring 11 that caused serious water turbidity problem during the horizontal cut by the bandsaw.

  While the blade cut off the 'teeth' in the vessel flange, it had also to cut through the packed[1] thermal insulation. Due to the moving of the blade, a small part of this insulation went into the pool water decreasing the pool visibility as a result. This visibility became very poor after the horizontal cut with the movement of the remaining rings on the turntable (ring 11 and 10). A cleaning of the turntable and filtering of the pool water was necessary. As a matter of fact, we had the same problem as with the removal of the thermal insulation (see B.4).

  The removal of this thermal insulation and the filtering of the water took 13.5 shifts.
- Mechanical bandsaw problems

  In the preparation phase of the vessel dismantling, we revised the bandsaw. The changing of all the bearings and seals and also a global check up of the bandsaw were part of this maintenance. Nevertheless, a mechanical problem occurred in the beginning of the vertical cutting campaign.

  The drive wheel of the band saw machine is equipped with an inner gear bolted to the main wheel port. Some of these bolts broke at the beginning of the campaign. In a first attempt, one replaced these bolts by others of a higher quality, but the problem remained. After a close examination of the driving mechanism, the shaft of the hydraulic motor showed too much clearance, probably caused by a damaged bearing (4 years of utilisation), giving too many vibrations to the main wheel. Changing of this bearing solved the problem.

  In total, the solving of this problem took 4 shifts
- A last problem concerned the waste racks where the cut pieces were put after the cutting of the ring. The guiding pins of the waste racks (the cut pieces had to be put between these pins) were too fragile. This means that when such a pin was hit during the placement of the pieces, it bend. This caused a lot of problems with the further placing of the remaining pieces in the waste rack and there was even a great risk that the hitting piece fell out of its gripping tool. This sort of problems caused a great delay in the cutting operations of the ring. This problem is difficult to quantify but one example can pinpoint the problem: the waste manipulation of ring 10 took 5 shifts while normally this manipulation should ask less than 2 shifts.

---

[1] Note: The original drawings of the RPV indicated that the insulation at that place was "packed by hand".

➢ **Waste production**

During the vertical cutting a certain amount of waste was produced. This waste is normally split up in two groups, the primary waste and the secondary waste. One considered as primary waste the cut pieces of the vessel, and as secondary waste all the waste produced while using this technique. In this case the swarfs and the filters of the different pool filtration systems are considered as secondary waste.

Waste removal system

For the removal of the high and medium activated waste of the reactor internals dismantling (former operation), the team designed a removal system based on two racks where the cut pieces were stored and immobilized and a support structure on which the two racks glided. This system fulfilled the different Belgian waste conditioning and safety requirements.

Using this system, some shortcomings appeared:

- The weight of the racks and supporting structure was not optimised (as the total weight per drum is limited by the national radwaste agency).
- The manipulation of this type of racks was difficult.
- The gliding of the racks over the support structure was not always easy due to an asymmetrical position of the cut pieces in the waste rack.

For the dismantling of the RPV, the team redesigned an improved system (**see figure 41**). The layout of the waste rack itself remained, but a bolting system replaced the support structure. Four bolts were placed in the tubes of the upper rack and a threatened hole was put in the tubes of the lower rack. A special long handling tool was designed to bolt the two racks together. Beside that, the team carried out a renewed mechanical resistance study in order to minimize the weight of the waste rack. The improved design of the waste removal system had thus 2 main advantages:

- less weight than the old system and so a higher effective waste load;
- just one, already existing lifting device was necessary to carry out the different rack manipulations.

This waste removal system with the cut pieces was then put in the waste transportation container (**see figure 42**) for further waste conditioning and intermediate storage at Belgoprocess.

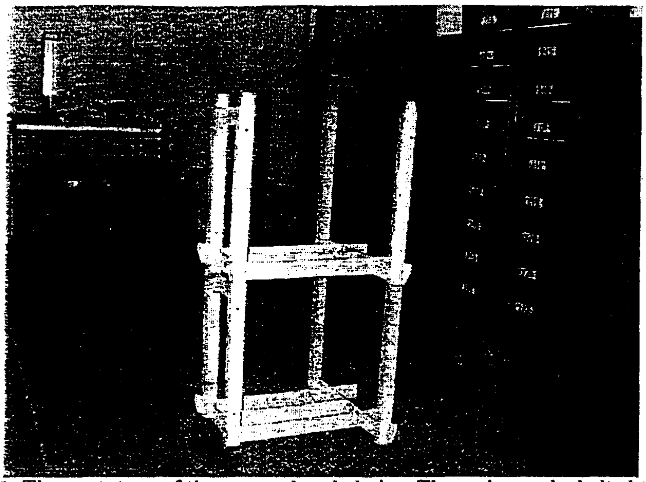


Figure 41: The prototype of the renewed rack design. The racks can be bolted together and are manipulated with only one lifting device.

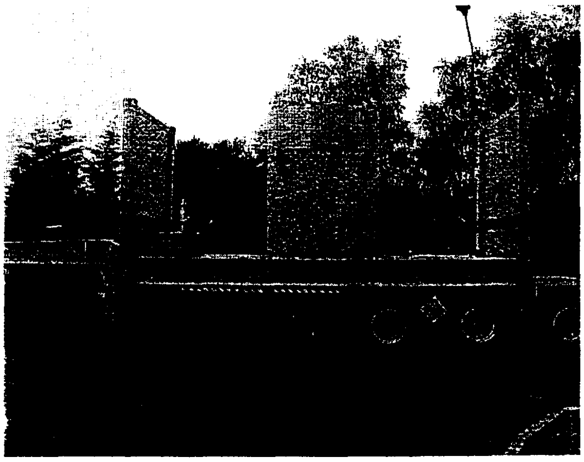


Figure 42: View of the waste transport container for the high level radioactive waste

Primary waste

The dismantling of the RPV produced solid waste in the three Belgian waste categories; high, medium and low activated metal waste (or HLW, MLW, LLW). The table below represents the quantity of the produced waste types.

Table 19: Quantity of produced waste types

Waste type	Quantity (tons)
HLW	5.15
MLW	7.22
LLW	10.5
total	22.42

Remark: The vessel head and bottom and the insulation shell are not included in the table above. They will be cut with another technique in the next phase of the dismantling project.

Secondary waste

The quantity of swarfs forms a first group in this type of waste. The calculated weight of produced swarfs was 357 kg. The amount of swarfs was also distributed over the three waste categories. The table below compares the calculated weight with the measured weight of swarfs.

Table 20: Difference between calculated and measured weight

	Calculated (kg)	Measured (kg)
HLW	172	87
MLW	54	108
LLW	131	162
Total	357	357

The HLW-swarfs are collected in special designed waste baskets which are compatible with the renewed waste rack design (**figure 42**). The great difference between the estimated mass of high activated swarfs and the removed mass to the waste conditioner can be explained: first of all, one ring was estimated as high activated, but in reality this ring was medium activated. This means that the estimation was already too high.
Secondly, as the difference in density between the swarfs and the base metal can be as high as 8 (1 g/cm³ swarf density; 8 g/cm³ base metal density), the specific activity of the swarfs can be reduced by the same factor. Therefore, the swarfs originating from metal being near the lower limit of a category can fall into the lower category only by this density difference. The above reasoning is also valuable for the other waste categories.

The high activated waste will be transported to Belgoprocess, the Belgian waste conditioner, with 9 transports (primary waste and high activated swarfs) representing 9 400 l drums or a waste volume of 3.6 m³.

The filters of the different pool purification installations form a second main group. First we have the original and main pool filtration unit where the filter consists of 64 cartridge filters of 10". Second there is the auxiliary pool filtration equipment were the filter consists of 5 cartridge filters of 30" and two ion exchangers (see also paragraph B.2.3.). **Table 21** below gives the volumes of these filters for the different main steps as well as the volume on worn tools for both techniques.

Table 21: Volumes of filters and worn out tools

	Main pool Filtration (dm³)	Auxiliary pool filtration (cartridge) (dm³)	Auxiliary pool filtration (ion exchangers) (dm³)	Tools (dm³)
Horizontal cutting	184	12.15	100*	8.8
Vertical cutting	184	36.45	100*	11.8
Total	**736**	**73.2**		**20.6**

*The same columns were used for horizontal and vertical cutting so that no difference could be made between the two.

➢ Comparison between both cutting techniques

Two different mechanical techniques were used to dismantle the same piece. Therefore it is interesting to make a comparison between both techniques.

To have a relevant comparison, only the cutting of similar rings was taken in to account. This means in practice that the values gained during the cutting of the reactor flange are not taken into account in the comparison. Also the workload and the received collective dose related to the visibility problem during the use of the bandsaw, are not calculated in the compared data.

To compare both techniques, all the data of the circular saw are recalculated to a cut section equal to that of the bandsaw (430 dm^2). This means dividing the circular saw data with a factor 1.026. This recalculation gives us the opportunity to calculate the ratio circular saw (CS) and bandsaw (BS) that is given in the 6^{th} column of **table 22**.

The table gives also the ratio calculated after the internals dismantling to show the influence of the piece thickness (vessel: 112 mm compared to the mean 25 mm of the internals).

Table 22: Comparison of the data of both techniques

	Unit	Circular Saw (gross values)	Bandsaw	Circular saw Equivalent	Ratio C.S.eq/BS	Previous ratio (reactor internals)
Cut section	dm^2	441.5	430	430	1	1
Total cutting time (8h shift)	Days	69	63.5	63.3	0.99	0.99
Total workload	Man-hours	1908	1236	1858	1.5	1.05
Observed collective dose	Man-mSv	3.31	2.64	13.11	4.9	1.02
Kerf width	mm	6	2	6	3	3
Removed metal volume	dm^3	6.62	8.73	26.21	3	3
Number of used blades		9	26	36	1.4	0.49
Blade cost per unit	BEF				4.9	26.9
Total blade cost	BEF				6.8	13

The table tells us:

- The removed metal volume (secondary waste) is three times higher with the circular saw.
- The circular saw is a machine with a high blade cost.
- The circular saw is more labour intensive when the thickness of the piece increases. This is shown very clearly by comparison with the ratio of the internals. Indeed, cutting a thick walled piece asked much more time. We had indeed to execute several cutting passes with the circular saw and some introductionary cuts for avoiding a vertical drift of the blade.

Conclusions:

- The bandsaw is more preferable than the circular saw when the thickness of the piece increases, and when the bandsaw can be used.
- The bandsaw produces less secondary waste (swarfs and worn blades) due to a lower kerf width.
- The use of circular saw requires much more stiffness of the workpiece (to avoid vibrations) than the bandsaw.
- When both techniques are applicable, the bandsaw should be preferred but, however, the choice between both techniques will be mostly guided by the geometry of the work piece and the accessibility of the environment.

EWN

The available reactor protection container will be used for the transport of the reactor components (except the reactor pressure vessel). Before the reactor components can be transported to the wet and dry cutting station, the basic frame with the shielding and handling system in the hatch area above the cutting station must be mounted. The reactor protection container with the reactor component will be placed with the 250 Mg crane to the ring of the shielding and handling system. By means of the internal lifting facility of the reactor protection container, the component will be lowered down on the turntable of the wet and dry cutting station.

Cutting and packing in the wet cutting station

The RPV-components will be cut from top to bottom. During the cutting, the components are fixed installation on the turntable in the cutting pool and covered with water.

The cylindrical coat parts of the core basket, reactor cavity and cavity bottom are cut with an underwater band saw in the cutting pool. Holes with a diameter of about 70 mm are drilled into defined spots. First, vertical cuts are made into the cylindrical coat. Then, the reactor component is rotated with the turntable in a certain angle and further vertical cuts are made. The last vertical cut ends in the drilling made before.
The saw blade is rotated into horizontal cutting position (90°). By driving out the horizontal carriage, the pre-cut segments are cut. The segment is held by the packing manipulator and after cutting under water put into the waste basket. The waste basket can accommodate several cut parts. The loaded basket is drawn into a shielding bell with sliding bottom. For this purpose, the bell is positioned above the water surface. After the waste basket is drained, the sliding bottom is closed and the shielding bell is transported with the crane to the packing station. After the sliding bottom and a shielding device is opened, the basket will be put in the filling box into the container. The weighing device in the filling box controls the loading process. When the container is loaded, the shielding door to the lid-closing station will be opened, the vehicle with the container goes to the lid-closing station and the door will be closed again. The lid is remotely put on the container. Then, the container lid is manually bolted or by the help of auxiliary devices. After registration of the radiological data, the container is transported to the storage place.

The wet cutting place is also designed for thermal cutting. In principle, the cutting process is similar to the mechanical cutting, i.e., all transport and preparatory works are identical, the cutting plan basically the same.

During the cutting process, the accumulated material from cutting is directly sucked off from the cutting point and collected in the filters of the water cleaning facility.

Specific cutting characteristics

In case of the protecting tube unit, the lower part (bottom plate) had direct contact with the fuel assemblies, thus, this part is more activated than the upper part (lower part 10^8 Bq/g and upper part 10^6 Bq/g).That's why it is intended to cut the bottom part of the protecting tube unit with a band saw in the cutting pool. During cutting, the protecting tube unit is positioned on the turntable and held by the lifting facility of the reactor protection container. After the cut, the lower part remains in the cutting pool and the upper part is drawn into the reactor protection container. The crane of the reactor hall transports the reactor protection container with the upper part of the protecting tube unit to the dry cutting station for cutting.

The reactor cavity and cavity bottom of the reactors 1 - 4 are tightly connected by wedges and bolts. As the cavity bottom had to accommodate a part of the fuel assemblies during reactor operation, the activity is higher (10^8 Bq/g) than in the upper part of the reactor cavity (10^4 Bq/g - 10^6 Bq/g). That's why, also here, a horizontal cut is made. The cavity bottom with a part of the reactor cavity remains in the cutting pool and the upper part of the reactor cavity will be cut in the dry cutting station. The reactor protection container is used for transport.

Cutting in the dry cutting station

The reactor components are cut from bottom to top. In the pre-cutting station, rings will be cut with horizontal cuts. To balance the weight and for lowering the reactor components after cutting, the load is taken over by a flexible wire hoist.

For the cutting process, the band saw is driven from the intervention area to the pre-cutting area. The cutting is automatically performed and monitored from the control room by a camera.

When the cutting depth is achieved, the cutting device is driven out of the cut. The reactor component will be turned with the turntable. The draw spindle of the lifting and dropping device is pivoted so that the reactor component can be turned into both directions. The reactor component is turned with the turntable up to 180° and another cut is performed.

Due to weight balancing of the lifting and dropping device, the rings can be cut safely and in case the cutting device gets stuck, a further load balancing with the lifting and dropping device will help. If this will not be sufficient, cutting the saw blade can relieve the band saw.

The cutting device is driven back to the intervention area of the pre-cutting place and the shielding door to the cutting area will be closed. The cutting device will be prepared for the next operation. The transport vehicle with the turntable and the cut ring is driven to the post-cutting place.

Here, the cylindrical rings of the reactor component will be vertically cut. A packing manipulator takes the cut parts and transports them to the filling box. The fixing construction of the turntable holds the remaining part.
After cutting and packaging, the transport vehicles with the turntable and necessary fixing construction go back to the pre-cutting place. The dismantling steps will be repeated.

KRB-A

Dismantling of the pipes in the upper ring room

As the German final storage was foreseen not to be closed before July 2000, it was planned to dismantle a part of the Biological Shield prior the RPV. Therefore it is necessary to cut and remove the pipes, which are connected with the RPV in the upper ring room. This will also ensure, that the upper RPV segments can be lifted up to the post dismantling area. There are four primary steam pipes, two feedwater pipes, one core spray line and eight measuring pipes, which are foreseen to be cut without lowering the reactor water level (**Figure 43**).

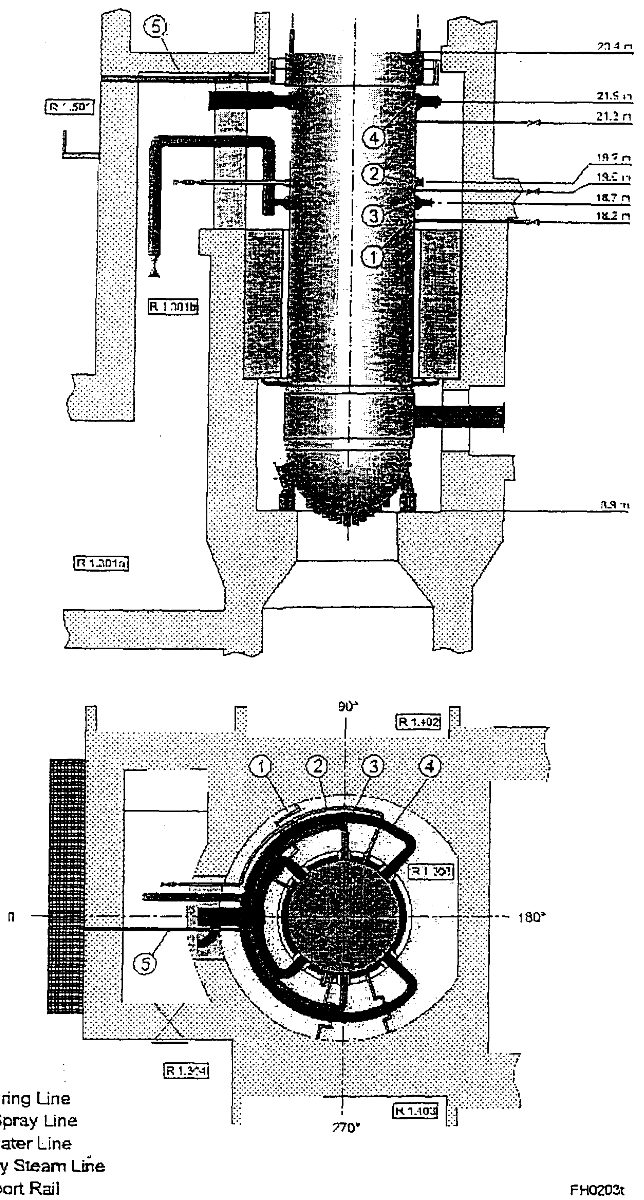


Figure 43: Position of the piping in the upper ring room above the Biological Shield

Cutting off the measuring pipes starts with icing the studs for sealing. Freezing will be done by a cooling jacket and liquid carbon dioxide. To drain the remaining water, holes must be drilled into the walls of the pipes. The pipes will then be cut off with a handhold band saw. The studs will be closed again by welding.

The icing principle is not suitable for the other pipes, because of their larger diameter. Therefore, sealing plugs have to be inserted into the studs of the primary steam pipes, feedwater pipes and the core spray line from inside of the RPV. This will be done by a device, which is carried by the reactor building crane. The device is controlled manually with rods from the refuelling platform. After draining of the pipes, cutting will be performed by several simultaneous and automatic working piercing saws. When the pipes are removed, the studs will be closed by welding on steel plates. The pipes must be post-dismantled for transportation. This is performed also by using ~~th~~ piercing saw~~s~~. Dismantled parts must be sealed with plastic caps before transporting in order to avoid spreading of contamination. The last step is to take the plugs out of the studs within the RPV.

All other components in the upper ring room had to be removed in preparation of this project. These were mainly the neutron flux measuring tubes, the shield cooling tubes on top of the Biological Shield and the RPV-installation supports (**Figure 44**).

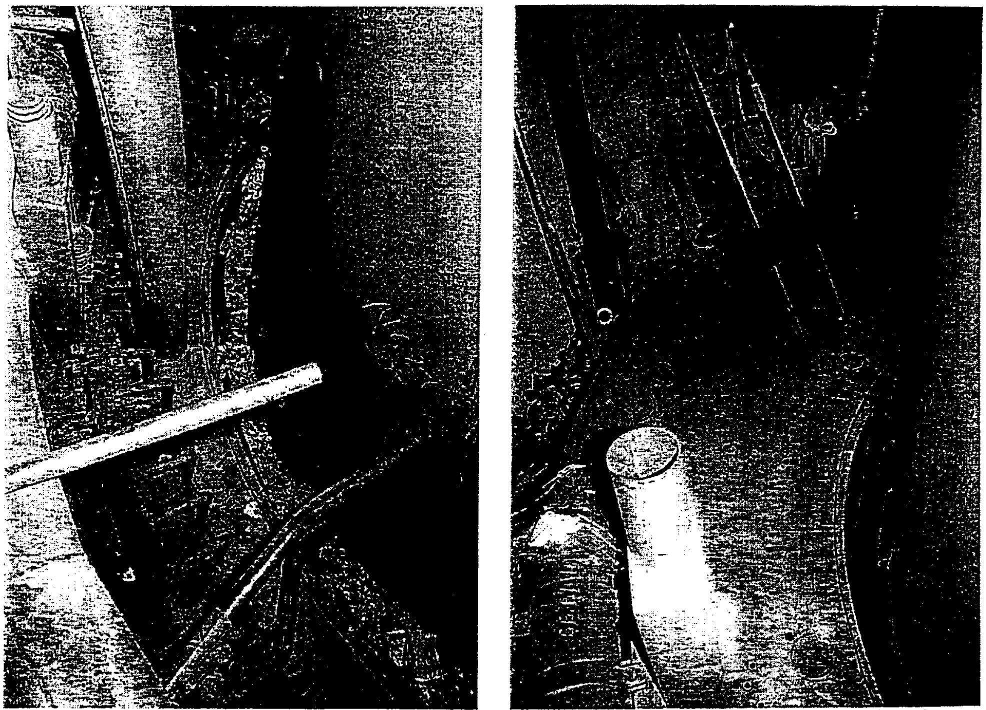


Figure 44: View to the upper ring room above the Biological Shield. Measuring pipes, cooling pipes and installation supports (left) were cut and removed (right).

The gap between the RPV and the Biological Shield was still filled with the insulation over the complete length of the core area. The insulation could manually be torn out from the lower ring room with long hooks and poles. This was an important requirement to get access to the RPV for the later cutting with a torch from the outside.

The working area around the RPV was cleaned intensively to reduce the hazard for incorporation of radioactive particles. Moreover the primary steam and feed water pipes were cleaned from inside by high pressure water jet and some hot spots in the pipes could be eliminated. The dose rate in the upper ring room could clearly be reduced by both measures.

Dismantling of the thermal shield and core shroud by plasma arc cutting

Prior to the dismantling of the core shroud, the first use of the unit had to be tested by cutting the low activated thermal shield. The technical experts, commissioned by the legal authorities, inspected the practical qualification of the machine during this cutting task.

The thermal shield is a cylindrical ring, 800 mm high and 6 mm thick. It is positioned closed to the RPV wall, directly beneath the condensate sparger ring. Handling and positioning of the dismantling unit could be tested in 15 m water depth, as well as cutting small windows for the load hooks. In opposite to the strategy of the core shroud dismantling, the condensate sparger ring had to be cut into several parts. A window was cut into every part, to insert a special hook for transportation. The position of all cuts is shown in **figure 45**.

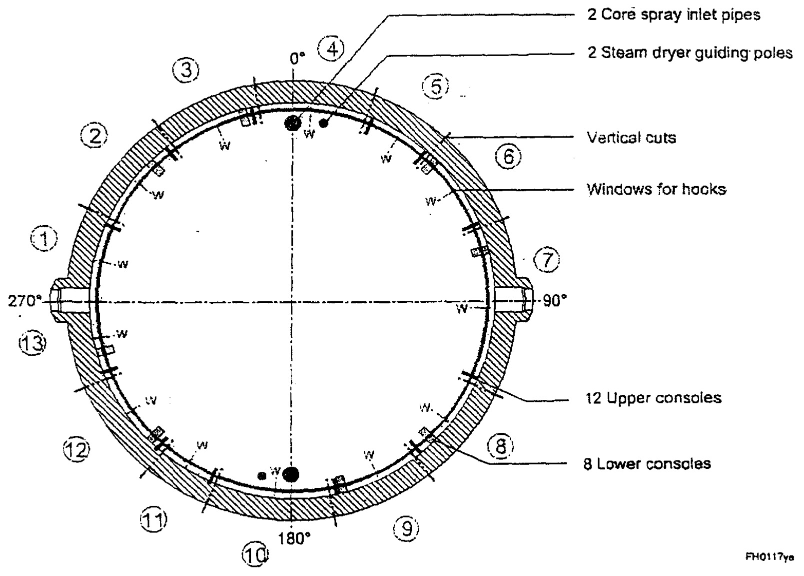


Figure 45: Position of all cuts for the thermal shield

After the thermal shield was cut and removed, the manipulator arm of the unit was modified to reach 6 stabilisers for cutting, which were positioned in the gap between the core shroud and the reactor wall. Moreover two core spray pipes and two guiding poles had to be cut, to disconnect the upper part of the core shroud completely from the reactor wall.

The first segment of the core shroud was cut right below the inner core spray pipes through a wall thickness of 25 mm. The cutting speed was set to 350 mm/min, considering a factor of safety. With a perimeter of 10 m the cutting process took about 30 minutes.
Afterwards the segment was lifted out of the RPV for post dismantling and packaging.
Figure 46 shows the first segment of the core shroud being positioned on the turntable in the post-dismantling area.

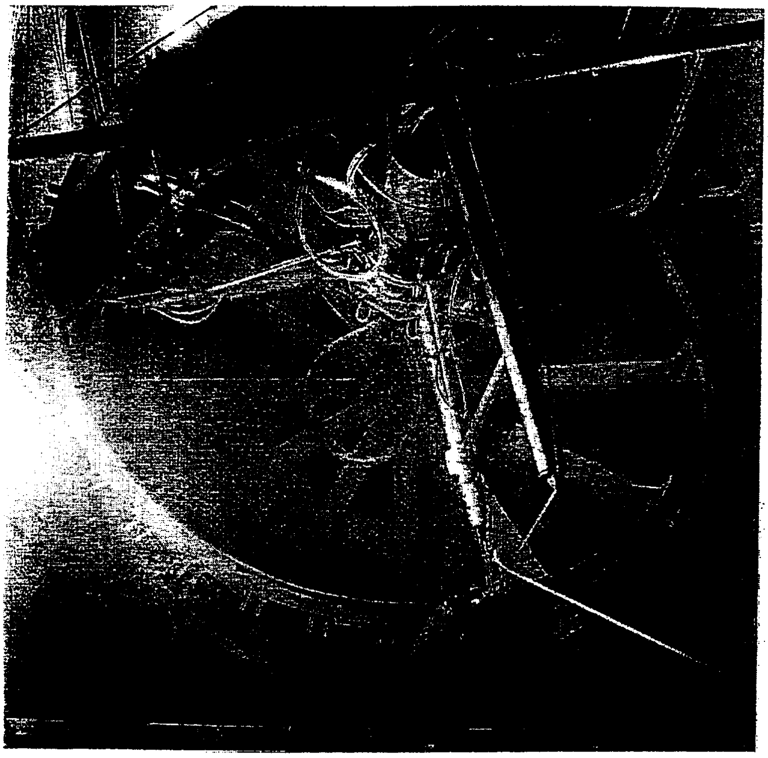


Figure 46: A segment of the core shroud is positioned on the turntable for post-dismantling

In preparation of dismantling the RPV, the inner walls have to be cleaned to avoid spreading of aerosols resulting from the contamination when using thermal cutting techniques. Also for this purpose the dismantling device for the core shroud will be used. In this case the manipulator arm will be equipped with a high pressure water jet tool instead of the plasma torch.

B.5. Collection of specific data

Specific actual dismantling technical data were collected in the three projects throughout the execution of the whole contract. The most representative ones will then be transmitted to the EC database.

BR3

The most important and best relevant data were collected during the actual dismantling of the reactor pressure vessel. Comparison could also be made between the cold testing on model and the actual work on the pressure vessel.
The data on the horizontal cutting by circular sawing and on the vertical cutting by band sawing, as described in section B.4, will be transmitted to the data base manager for integration.
Specific literature data and reports references will also be sent for integration.

EWN

Test of reactor component cutting

Transport of reactor pressure vessel

For the test of the reactor pressure vessel cutting, the reactor pressure vessel for unit 8 is used. It is not activated and not contaminated and was stored outside the technological facilities in a lightweight hall construction to be protected against weather influences.

In preparation of the transport to the steam generator room of unit 5, the nozzles for the reactor coolant pipes (diameter 500 mm) and for the core flooding systems and measurement connections were cut by thermal cutting techniques. For the reactors 1 to 4, these works are done in installation position with a milling facility.

With the transport to the steam generator room of unit 5 (**see figure 47**), the transport of the reactor pressure vessel of unit 5 to the Interim Storage North was tested. Hereby, it was proved that the transport way and the crane is suitable for the reactor pressure vessel transport of unit 5 to the Interim Storage. Due to the short operation time of reactor 5 and its low activity, the individual reactor components will be transported as a whole to the Interim Storage and will be manually dismantled and disposed of after a corresponding decay time.

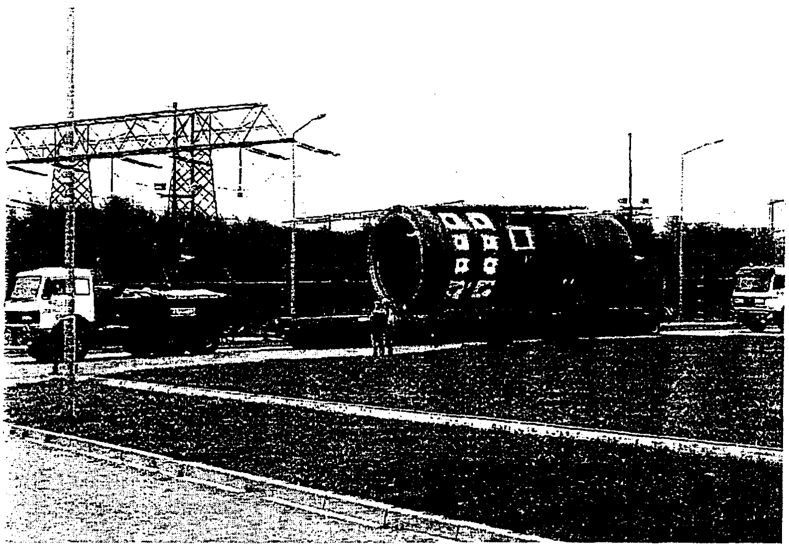
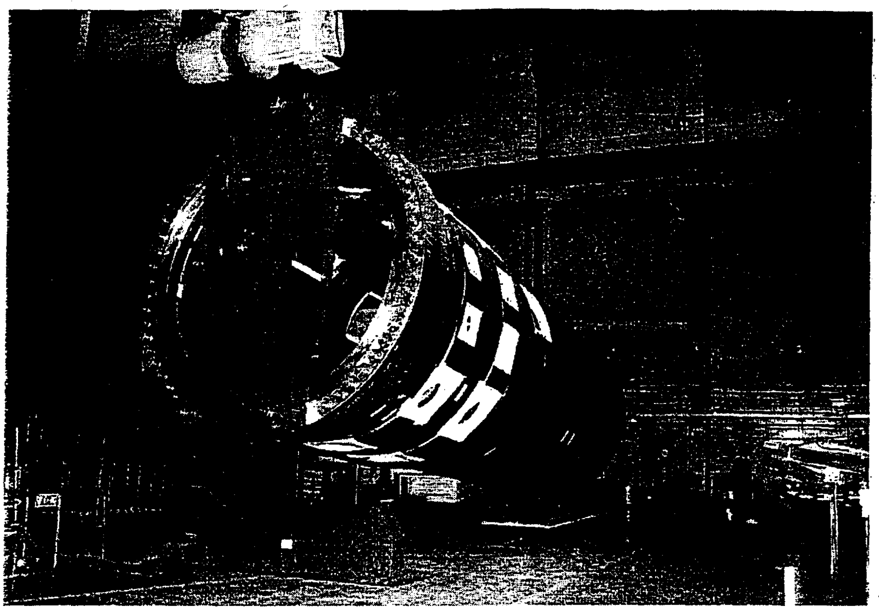
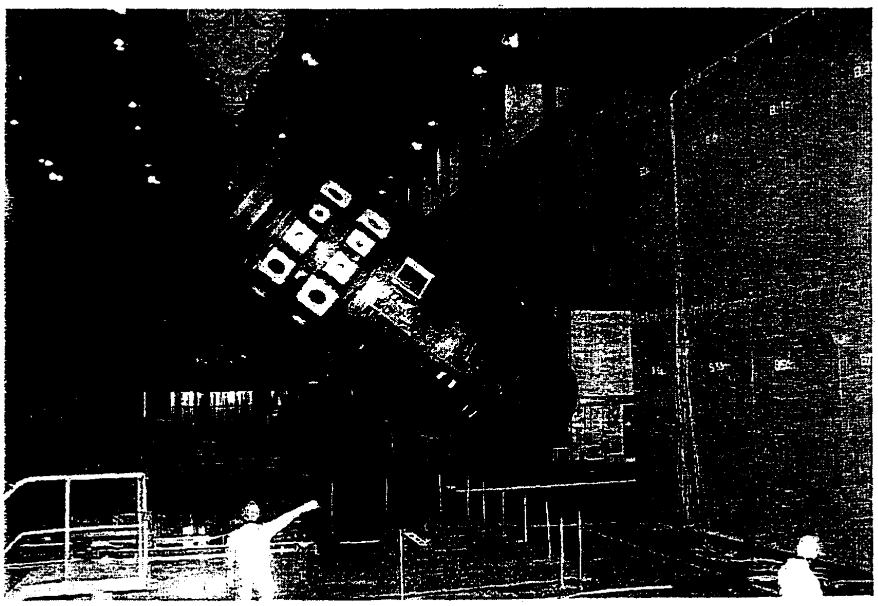


Figure 47: Transport of the RPV of unit 8 t unit 5

Afterwards, RPV 8 was transported to the controlled area of unit 5. Before the RPV was put to the dry cutting place, it was brought from horizontal to vertical position with a special tilting device (**see figures 48 and 49**). This was also a test for the later disassembling of the reactor pressure vessel 5, but then, the pressure vessel will be tilted from the vertical to horizontal position. On the basis of the test plan, the man-hours needed for the work were determined. On the basis of the known dose rates of the pressure vessel (in the flange area, nozzle area, core area, bottom), the average ambient dose rate for the individual activities were determined. For the removal of the RPV from its installation position and for tilting, a collective dose of 1.4 mSv was evaluated.
The RPV 8 was set down on a special holding device located in the upper hatch area and consisting of large bolts which remotely move into the traverse of the RPV. Thus, it is possible to set down the pressure vessel. For moving the RPV up and down, a wire hoist is used and a crane will not be needed.

The transport of the reactor pressure vessel and the positioning in the dry cutting place went according to plan without problems. To minimise the radiation exposure, the reactor hall crane and the wire hoist have to be improved concerning their remote handling ability when RDG 5 and the reactor pressure vessels 1 - 4 will be transported inside the reactor hall.





Figures 48 and 49: Tilting device for the reactor components

Reactor pressure vessel cutting

In the sequel of the tests, the reactor pressure vessel was put on the transport vehicle with turntable by the wire hoist. A load of 18 t is carried by the turntable, the remaining weight by the wire hoist (approx. 200 t).

The first step is the cutting of the domed reactor pressure vessel bottom (**see figure 50**). This will be done with band saw and oxyacetylene cutting. It is planned to mainly use the band saw and only when necessary oxyacetylene cutting.

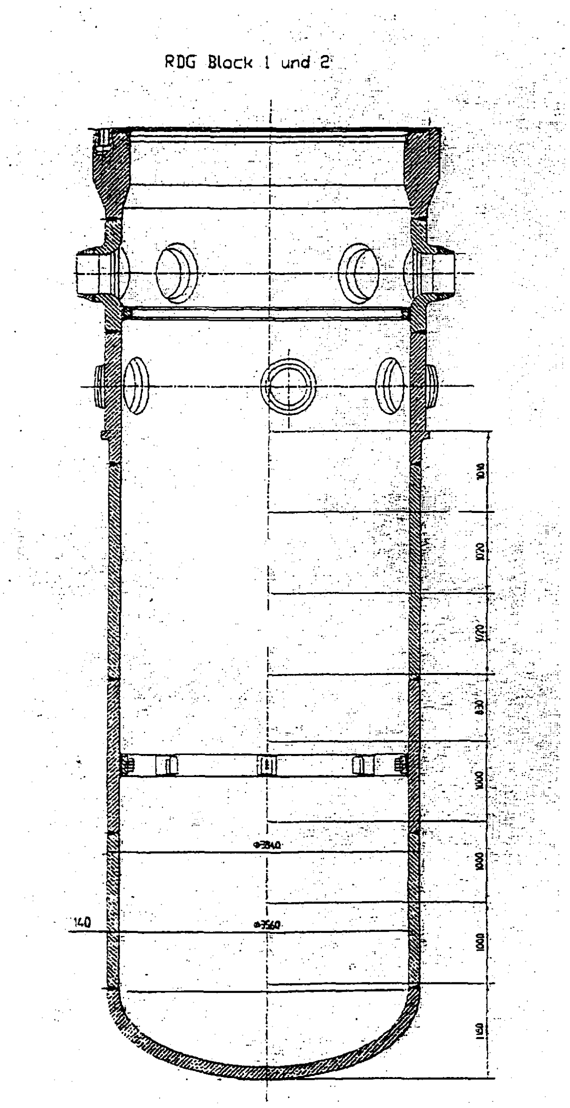


Figure 50: Cutting point of RPV

For the work performed until now, only the band saw was used (**figure 51**).

Figure 51: Cutting of the domed RPV bottom with band saw

By variation of cutting speed, cutting depth and tooth pattern of the sawing band, the band saw operation could be optimised. Great importance was attached to the optimisation of the produced metal shavings. As these shavings are sucked off near the production place and are routed to the ventilation system, small (maximum length 10 mm) and straight shavings are required. The shavings must not hook together to prevent plugging of the ventilation pipes. To optimise the shavings, the band speed was varied between 10 m/min and 70 m/min and the contact pressure of the saw band between 250 N and 700 N. To have an optimum shaving production, the saw bands should have the following parameters:

1st tooth shape:	claws	
tooth space:		1.4 up to 2 teeth per inch
band speed:		60 to 70 m/min
contact pressure of the sawing band:		ca. 600 N
2nd tooth shape:	claws	
tooth space:		2 to 3 teeth per inch
band speed:		40 m/min
contact pressure of the sawing band:		ca. 500 N

In parallel to the optimisation of the cutting parameters, the cutting geometry has also been optimised (rotation angle of the reactor pressure vessel, cutting depth, cutting length, number of cuts). The aim was to guarantee a safe band saw operation with as less as possible cuts (reduction of cutting time) and to make a continuous cut through the circumference of the reactor pressure vessel. The following parameters were used for optimisation:

Table 23: Parameters used for optimisation

Rotation angle of RPV (°)	Cutting depth of band saw (mm)	Cutting length (mm)	Number of cuts
14	175	801	25
36	250	947	10
51	325	1069	7
60	400	1173	6
72	475	1264	5

It turned out to be advantageous to use the ending cut for the positioning of the new one. Furthermore, it was found out that big cutting depth lead to an instability of the saw band between both cutting points and the cut is not horizontal. The works for optimisation of the band saw operation will continue with the cuts on the cylindrical part of the reactor pressure vessel. All in all, the applicability of the band saw for horizontal cuts on the reactor pressure vessel could be proved.

After the cutting of the pressure vessel bottom, the remaining part of the vessel was lifted with the wire hoist and put on the support bolt. The vessel bottom was transported with the transport vehicle/turntable from the pre-cutting place to the post-cutting place.

As it is not possible to saw up to the centre of the vessel bottom, the cutting in the post cutting area is foreseen to be done in two steps:

1. Segment cuts will be made into the vessel bottom by band saw (**figure 52**)
2. The cutting of the middle part will be done by thermal cutting procedures.

Figure 52: Segments cut into RPV bottom with band saw

After stabilising the hydraulic facility for the band saw control, segment cuts up to a length of 70 cm could be made into the vessel bottom.

For further cutting with thermal cutting devices, it is planned to use plasma torches and oxyacetylene burner. For both burners, manual handling and use of manipulators will be tested.

A test to cut out a circular piece from the centre with a manipulator controlled plasma torch failed (**figure 53**). The plasma torch could not cut through a wall thickness of 14 cm from the inside. The austenite plating was completely melted in the burner surrounding but the melting material sealed the cut again before the burner could go through. The tests was continued from outside with an oxyacetylene burner (**figure 54**). For cutting, a gas burner was used. With this technology, the most part of the pressure vessel bottom could be cut. For the central part of the pressure vessel bottom (in a radius of ca. 80 cm from the centre point of the vessel bottom), a suitable technology for cutting is still investigated.

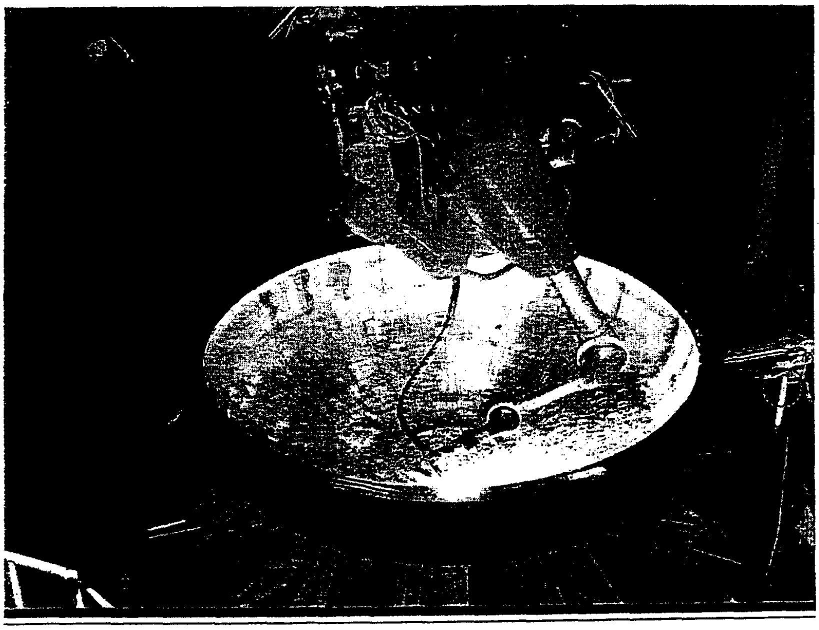


Figure 53: Plasma torch testing

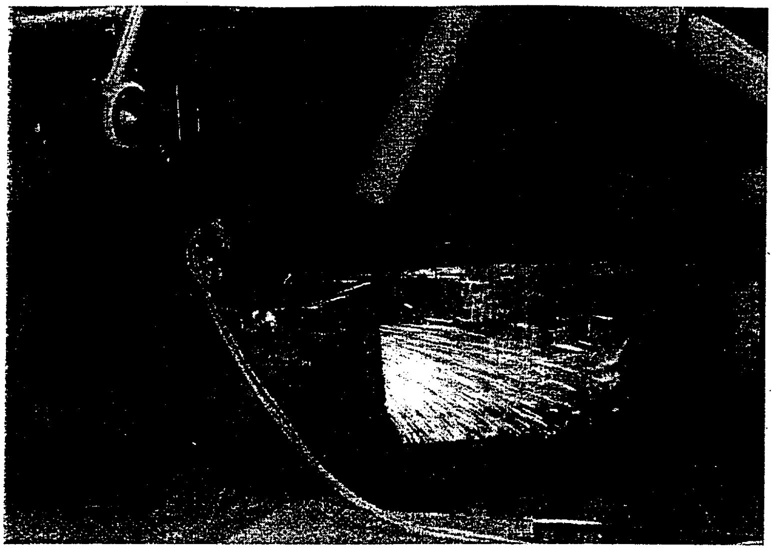


Figure 54: Cutting of vessel bottom with oxyacetylene burner

<u>Milling of the reactor pressure vessel nozzles</u>

To remove the reactor pressure vessel from its installation position, it is necessary to cut all nozzles from the outer wall of the vessel. Otherwise, it would be necessary to remove bigger quantities of concrete above the pressure vessel.

For the removal of the nozzles, a milling facility was developed. For the test, the completely installed but never operated pressure vessel from unit 7 was used. In June 1999, the milling facility was brought into the vessel (**figure 55**) and it was started with the cutting of the nozzles. Due to the difficult accessibility to the nozzles, the remotely controlled device was brought into the reactor pressure vessel and the nozzles were cut from the inside to the outside as in the BR3 reactor. For that, the device was suspended on the upper side of a carrying and shielding plate 200 mm thick which is put on the RPV flange and air sealed. On the part below, the device is fixed with 3 hydraulically movable tightening cylinders which are staggered by 120°.

Figure 55: Moving of the milling facility into RPV 7

The milling module consists of the guide block driven via three axes and the milling spindle with milling head driven by an electric motor. Due to the wall thickness of the nozzles of 60 mm and a diameter of 780 mm, several tools are needed for the cutting of a nozzle. Therefore, the machine was equipped with a store and a changer device for the used milling discs. Thus, the milling facility is provided with a changer magazine that can accommodate 3 milling discs. Through an additional opening in the carrying plate, a gripper with holding rod can be inserted from the outside and provide the milling module with 2 further milling tools.

The tests were performed to prove the functionality of the milling facility and optimise the milling process (**figure 56**). During the tests, heavy vibrations occurred also at the milling module. They could be partly eliminated by changing the tightening cylinders. Nevertheless, extensive test measures have to be performed to guarantee a safe milling operation. By optimising the milling parameters, the time for milling a nozzle could be reduced from approx. 100 hours to 11 hours.

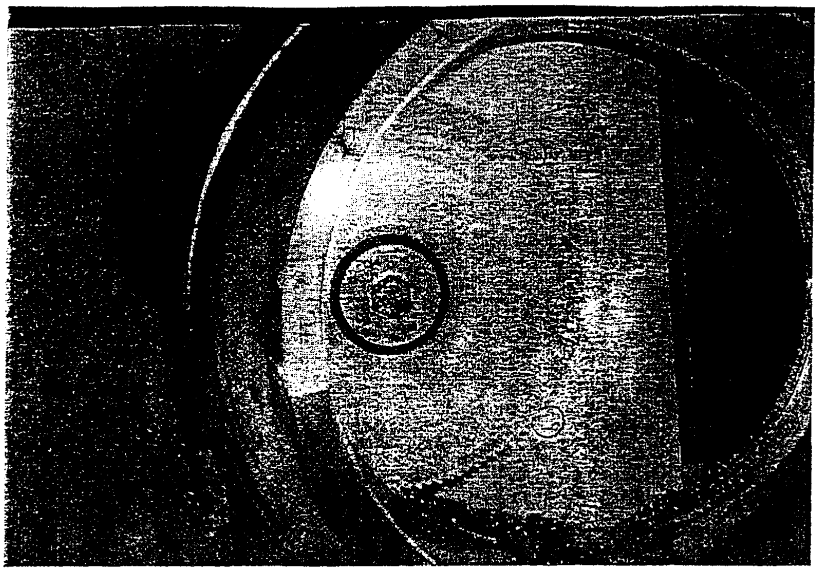


Figure 56: Cutting of a reactor coolant pipe nozzle

Cutting works of the core basket

The start of the transport and cutting works of the core basket have been delayed resulting from necessary improvements of the fixing and holding facility on the turntable of the wet cutting caisson. The dismantling of the cylindrical parts of the core basket started in December. The cutting plan is seen in **figure 57**. First, vertical cuts (**figure 58**) of approx. 840 cm length are made with the band saw from the upper edge of the cylindrical part of the core basket. After each cut, the core basket is turned with the turntable by 6° to 12°. After the vertical cuts are made, an opening of approx. 80 mm diameter is burned out with a remote CAMC device or plasma torch at the end of the cut. Then, the band saw is positioned at the end of the vertical cut and the sawing band is turned in the opening by 90° and a horizontal cut can be made. While cutting, the manipulator holds the part which has to be cut. This procedure is repeated for all other rings according to the cutting plan, see figure 15, until the cylindrical part of the core basket is dismantled.

The test cuttings performed until now have shown that the operation of the band saw has to be stabilised (hydraulic facility). Problems also occurred with the CAMC facility concerning the holding device of the graphite electrode. There were nearly no problems with the plasma torch.

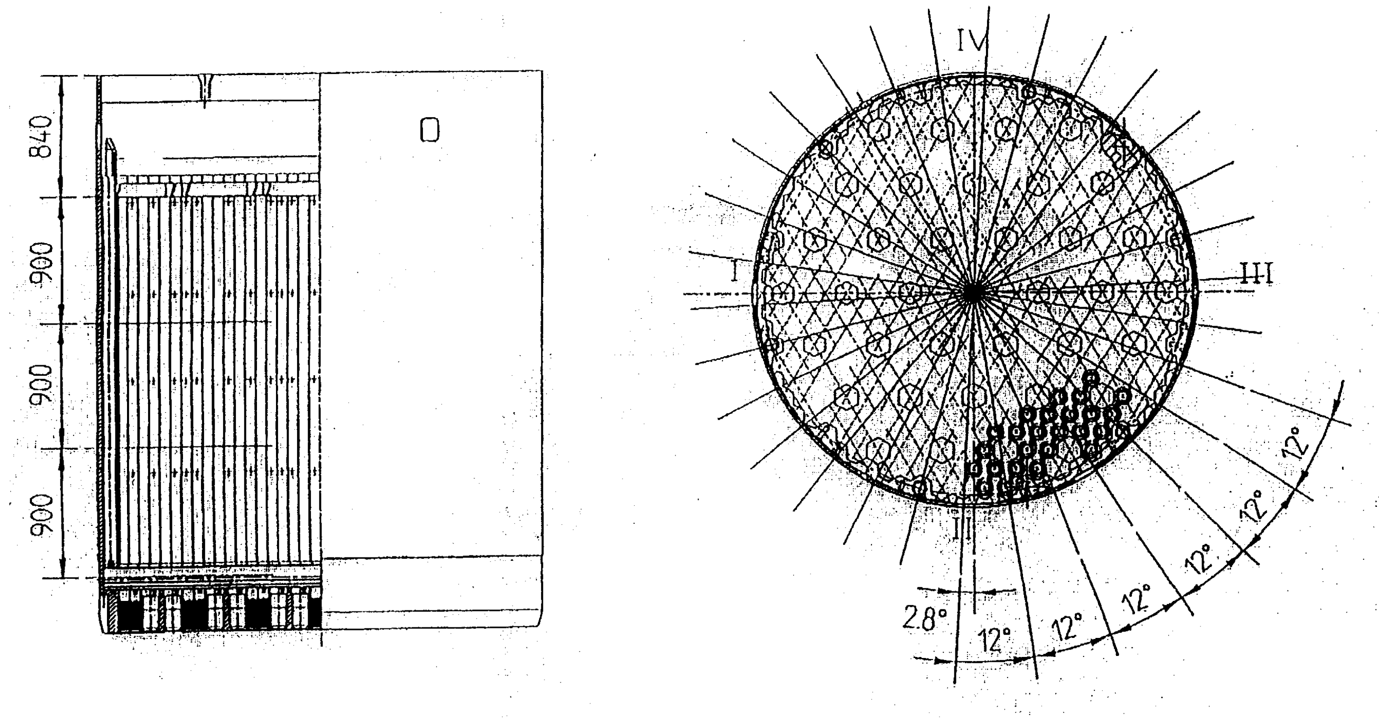


Figure 57: Cutting plan for the core basket

Figure 58: Cutting of core basket with band saw

4. TIME SCHEDULE

Within the three projects involved, some delays were encountered for different technical and administrative reasons.

Therefore, a request for schedule extension of 9 months of the project has been issued to the European Commission, and the Commission agreed on it, setting the project completion date on 30 April 2000, instead of the original 31 July 1999.

5. CONCLUSIONS AND MAIN RESULTS

BR3

The complete remote dismantling of the BR3-PWR pressure vessel proved the actual feasibility of such an operation. The selected strategy consists in removing the vessel in one piece into the refuelling pool for further segmentation.

The advantages were:
- Accessibility of the RPV insulation shroud from the outside. The works required for the separation of the shroud had to be adapted in order to cope with two unforeseen events: the fastening screws were not visible and working nearby the thermal insulation led to an important water turbidity problem. The outside accessibility with a simple equipment allowed some "flexibility" in the operation to solve the first problem. The second one was solved by additional purification equipment. Both problems delayed the project.
- Accessibility to the thermal insulation after the shroud removal. Again, this operation led to water pollution. Nevertheless, no insulation particles (powder) were spread out in the air. This is one of the most important advantages of working under water.

Both problems mentioned above were envisaged during the study phase but no actual experience was available and this is probably the reason for having underestimated the possible negative consequences.

- Great simplification for the further vessel segmentation. The "re-use" of equipment already used for the dismantling of the internals:
  1) avoids development/building of new equipment (time and cost)
  2) reduces the learning/maintenance time (first learning process factor)
  3) avoids additional radwaste
  4) allows to concentrate the cold tests on mock-up on the new required cutting parameters and clamping equipment.

  In this case, the learning effect has two positive factors: shorter cold tests and reuse of same well know equipment.

The selected strategy required to separate the RPV from the primary loop and to reinstall the water tightness of the refuelling pool. Both operations required high skilled engineered solutions. The first one was the development of a quite complex cutting equipment. Again extensive cold tests allow to test the reliability of the solution, to determine the operation parameters and develop alternative solutions in case of failure.
The second one was cold tested a first time. The design was then reviewed for a second cold test. Nevertheless, the actual operation had to be stopped. Non-conformities between

drawings and reality were observed. This led to a second design review. Finally, the operation could perfectly be carried out but the project was delayed and the dose uptake increased.

As general conclusion:
- quite simple mechanical tooling adapted for working remotely and under water can be used for most of the operations;
- due to the selected strategy important time was spent on the preparation and pre-segmentation operation;
- cold tests on mock-up are required; for the whole project it saves doses, time and money;
- external parameters and non-conformity between drawings and reality led to the main difficulties.

Nevertheless, the main operation was carried out safely, in an economical way and keeping the objective of waste volume minimization.

This project completes a process started with the dismantling of the reactor internals, and which has led to compare, test and use different remote dismantling techniques for segmenting the most radioactive part of a nuclear power plant. Although the BR3 plant is a low rated plant, the pressure vessel geometry (thickness) and material can be of interest for commercial size power plants.

EWN

Although the model dismantling has not yet been terminated, it can be concluded that the techniques and procedures selected are suitable.
Delays occurred in the commissioning of several equipment, notably dry band saw, video technique and packing station. This was in all cases due to deliveries of equipment non-conform with specifications.

The preliminary results from the tests have furthermore verified that:
- air and water cleaning systems are properly functioning during operational conditions;
- the emissions (air and water) are magnitudes below the licensed limits;
- the dose commitments evaluated for the different tasks until now are close to the planning values.

KRB-A

The dismantling of the internal components could almost be finished during the project. One of the main results was obtained by cutting the upper grid plate, which was the most activated part of the core support. Plasma arc cutting was the method of choice already for the steam dryer system. Therefore it was also chosen for cutting the upper grid plate. It turned out, that the secondary emissions, produced during the process, could be controlled safely. Special attention was spent on the radioactive aerosols, which have to be captured at the water surface. As expected, aerosol radioactivity could be measured in the exhaust duct, being distinctively higher than in former cutting procedures. Extensive studies were made, showing that no higher activity could be found after the filtering device or in the working area. No special respiratory equipment was necessary for the staff on the reactor floor.

As a consequence from this results it was decided to use the plasma technique for the core shroud too. A new manipulator was constructed, as the core shroud - in opposite to the steam dryer and the grid plate – had to be dismantled within the RPV. Having no restoring forces, the design of the manipulator could now be very lightweight. Another advantage is that the positioning of the plasma torch must not be that precise as for mechanical milling tools. The combination of this unsupported manipulator with a flexible thermal cutting tool turned out to be reliable in use, simply to handle and fast in cutting. Fixed to the crane of the reactor building, positioning of the manipulator is quite easy and the set up time remains short. Cutting in a water depth of 15 – 20 m did not cause any problem. The complete cut-off-time for a ring segment, including positioning and transportation to the post-dismantling area, could be managed in about four hours. Due to the complex structure of the first ring segment, the procedure of post-dismantling established to be more time consuming. However, it was decided to cut the segment, including two inner core spray pipes, into small parts, as the cutting speed of the plasma torch is very high. This extra expense in cutting is more than compensated with the resulting packing density and costs for the waste containers.

The strategy for dismantling the RPV in KRB A is taking into account the experience of cutting the RPV head. This project was finished during the first decommissioning phase, using thermal cutting tools. This technique is still preferred from the economical point of view, because mechanical tools ask for heavy and expensive support systems. In one of the dismantling studies it is proposed to use thermal cutting for the upper and lower part of the RPV, and to cut the high-activated core area with mechanical tools. But the actual strategy is to cut the core area in one piece and to post-dismantle it under water in the fuel element pool also by thermal techniques. A main requirement was to get access to the RPV from the outside, which was realised by removing the insulation between the RPV and the Biological Shield. Because of this, a gap of about 250 mm was opened, providing the opportunity to insert a small cutting tool. Following this strategy, it is possible to reduce the costs for tool support and equipment to a minimum.

The unexpected closure of Germany's final storage ERAM in Morsleben was very unpleasant for the project. Several projects had to be stopped although the final planning was finished or tools, components and systems had already been bought. One of the projects was the cutting of the pipes at the RPV in the upper ring room above the Biological Shield. It was intentional to get access to the Biological Shield for dismantling and removing the main part of it, according to the scheduled closure of the final storage in July 2000. Due to the new situation, this project became pointless, taking into account, that the dose rate in the working area will be lower when the dismantling of the RPV has been finished.

The time consuming problems with the water turbidity, which are described in the report, stressed the necessity of changing or extending the chemical analysis. Bacterial pollution of the water seems to be a fundamental problem during decommissioning, because the antiseptic effect decreases with the fading of radioactivity while dismantling the high-activated components. It is absolutely advisable to treat the water early enough with adequate measures.

Influenced by these unexpected changes, the time schedule of the whole project had to be stretched. In association with the problems of water turbidity the dismantling of the RPV suffered a delay of more than one year. The actual planning is, to finish with the dismantling of the internal components until the end of the year and to start with the dismantling of the RPV in 2001.

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DECONTAMINATION REPORT

Dismantling file	Infrastructure
Doc. nr.:	INF-OVE-28 118/454-01
Page:	7 (including annexes)
Date:	2002-- 10 – 10

Title:

MECHANICAL DECONTAMINATION OF A CONCRETE BLOC WITH DIAMOND DISC

This document was originally edited in English.

		Date	Signature
Author:	Fabbri Silvio	2002 - 10 – 10	
QA co-ordinator	Demeulemeester Yves		

1. Introduction

This report describes the decontamination of a concrete bloc, by means of manual tools with a diamond disc, following the work instruction 117/BR3/VS-454. The concrete bloc was used as a platform for two pumps placed in the control area and they were slightly contaminated on some surfaces. Their decontamination was achieved by removing 5 mm of material by using diamond tools on disc sander type Flex. The work was done in a ventilated and filtrated enclosure. The filtration group had prefilters that can be regenerated. The decontaminated bloc is evacuated as recyclable industrial concrete.

WAC number	4540
Task number	G023043

2. Results

The decontamination work was successfully done. The total time to perform the work was 4,5 hours. The decontamination team was composed of 3 persons.
The concrete bloc before decontaminated weight was 1910 kg included an iron plate of 172 kg. The total collective dose was 9 man.μSv.

3. Object to be decontaminated

The contaminated concrete bloc, used at BR3, was first measured by the Radiation Control (RC). The results gave contamination of less than 5 Bq/dm^2; the measurements were done by using the Selectra.
The concrete bloc covered with an iron plate was on pallet in the Frisomat. It was brought to the machine hall using a Clark were a crane lifted it to the workshop level.

4. Description of the works carried out

4.1 Preparation of the work place

➢ Tools

- Disc sander Flex with diamond disc ∅ 125 mm and protection head + spare disc.
- Disc sander Flex with special diamond tool ∅ 70 mm for throat.
- Aspirator Nilfisk and Nedermann of large capacity.
- Worktable + reservoir with a border of some cm.

➢ Workplace

- The mechanical decontamination of the concrete bloc was carried out in a ventilated booth, mounted in the machine hall.

4.2 Nuclear and classical security measurements

- All tools or material were evacuated of the cutting workshop.
- The Donaldson ventilation already worked.
- The appropriate protection means were available: paper overall, Gridel, etc.
- This technique produced a lot of dust inside the cutting workshop so all the tools were cleaned and measured before they left the workshop.

4.3. Decontamination

4.3.1 Team composition

- 2 decontaminators inside the workshop.
- 1 supervisor-SCK outside the workshop.
- 1 part time responsible for the work floor.

4.3.2 Execution

- A pallet with the concrete bloc was taken in the Frisomat and brought to the machine hall were a crane lifted it to the workshop level.
- The concrete bloc was introduced inside the cutting workshop.
- Before starting the ventilation the inner level of the drums to collect the dust had been verified to know the exactly weight collected.
- The pressure drop avec the prefilters and filters had been noted at the beginning and at the end of the work to verify the correct function.
- Before the decontamination the iron plate was removed manually from the concrete bloc and put it on the worktable.
- The operators treated the surfaces of the bloc and plate with the diamond disc.
- At the end of the work, one waits until the dust was evacuated from the workshop, then the door was opened and the concrete bloc cleaned and transferred to the C1 measurement.

4.3.3 Cleaning of the work space

The cleaning was already done when the concrete bloc was treated.
The tools were cleaned, aspirated and blown through with pressured air (in the cutting workshop). The workshop was cleaned completely. The dust drum was evacuated and weighed. The tools were checked and ordered.

4.3.4. Evacuation ways

The liberated bloc (1710 kg) will be sent after the acceptance of the clearance report as industrial waste. The dust (28 kg) drum was sent as A17, whatever their activity level.

4.3.5. Material flow

The numbers of the produced batches are the following ones:

BATCHES NUMBER	IDENTIFICATION	WEIGHT
BR3-02-009-B	CONCRETE BLOC	1710 kg
BR3-02-033-P	IRON PLATE	172 kg
BR3-02-061	DUST	28 kg

TOTAL: 1910 kg

5. Dose estimated and real

The ambient dose rate amounts to 2 μSv/h.

Operation	Number of persons	Number of hours	Dose (μSv /h)	Total
Preparation	2	1	2	4
Execution	3	1,5	2	9
Clearing	1	2	2	4
Total				17

The total dose estimated was: 17 man.μSv.
The total collective dose was 9 man.μSv.
The total hours to perform the work was 4,5.

The position of the people to perform the work makes the differences between the real and the estimated collective dose.

6. Lessons learned

The diamond disc grinder works well to remove decontamination surface in concrete blocs. Since the operators worked at the same time and produce much dust that sometimes is difficult to reach the clarity, one possible solution could be to install a bigger aspirator over the concrete bloc to be decontaminated.

7. Annexes

Photos 1, 2, 3, 4, 5, 6, and 7

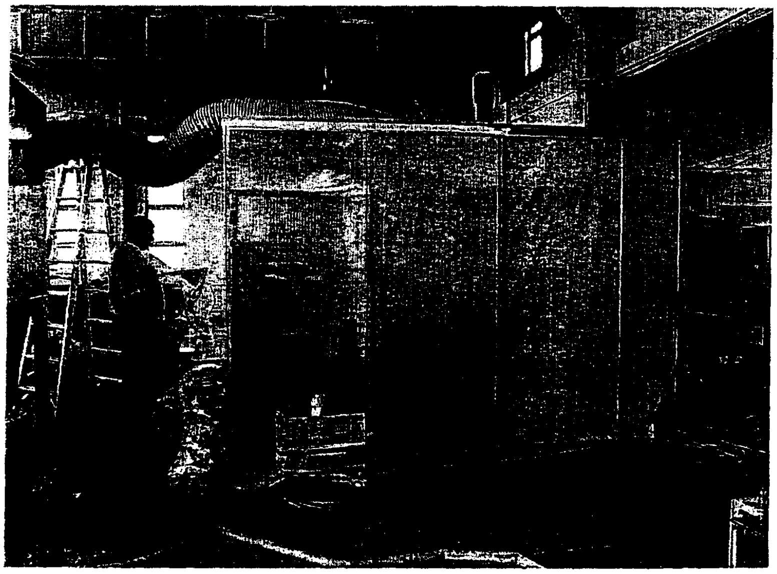


Photo 1
A view of the workshop in the machine hall

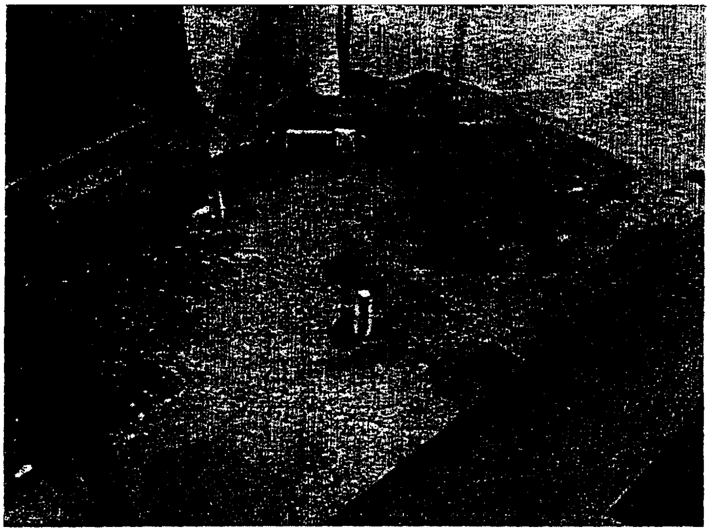


Photo 2

The concrete bloc and the iron plate before decontamination

Photo 3
Operators inside the workshop before the decontamination with the grinder

Photo 4
The concrete bloc after decontamination

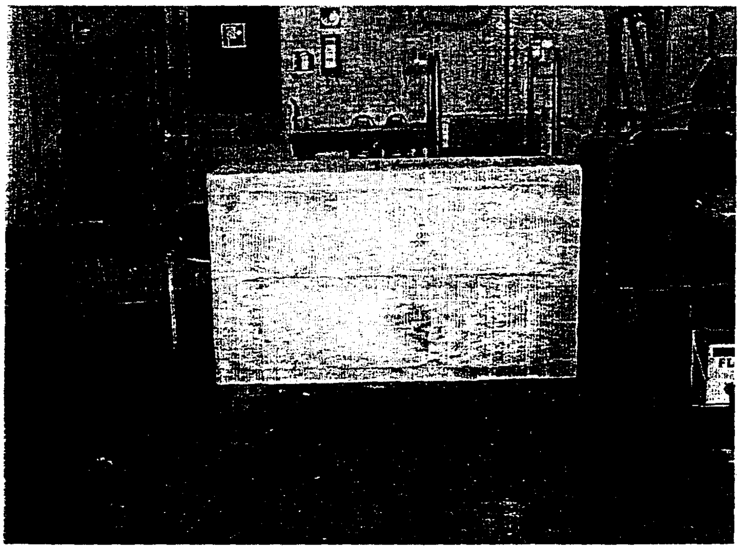


Photo 5
The concrete bloc on a pallet in the machine hall

Photo 6
Batch source identification over the concrete bloc

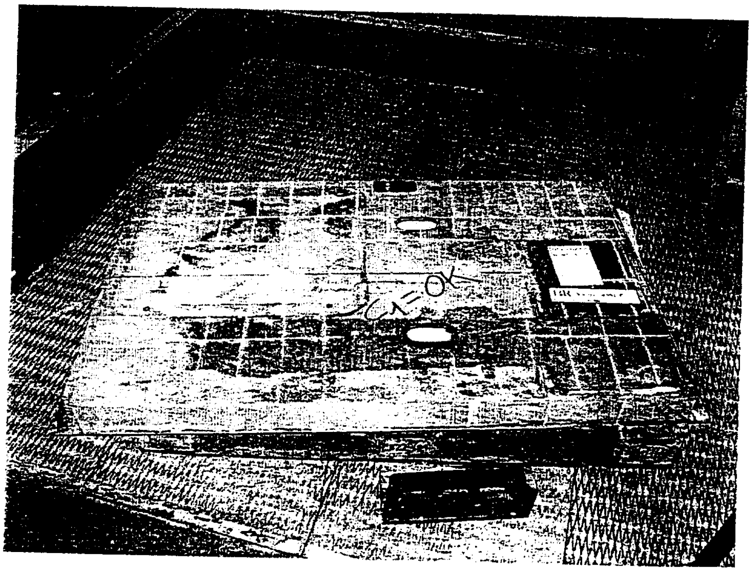


Photo 7
Iron plate cleaned

附件三

SCK 內部報告：5 份

- The BR3 Dismantling Operations and Related Techniques－Dismantling of the Reactor Pressure Vessel
- The BR3 Dismantling Operations and Related Techniques－Dismantling of Highly Radioactive Reactor Internals
- Waste Management at BR3
- Waste Management－Study of Dismantling Strategy
- Evacuation of the BR3 Spent Fuel

The BR3 dismantling operations and related techniques

Next

Dismantling of the Reactor Pressure Vessel

The presented phase of the project concerns the dismantling of the reactor pressure vessel (RPV).
The pressure vessel is a 28 tons carbon steel (ASME SA 302 gr.B Ni Modified) forged piece clad with stainless steel (ASME SA 240 gr.S).
Different strategies for the RPV dismantling were studied and led to the selection of underwater cutting, the RPV being removed from its pit into the refuelling pool.

- *The pressure vessel had then to be decoupled from its primary loop and unfastened.*
- *The refuelling pool leak tightness had to be reinstalled.*

This led to the RPV removal, carried out on the 24th of August 1999. The real dismantling of the RPV could then proceed. The thermal insulation and insulation shroud were completely removed in December 1999 and from January 2000 till June 2000, the main RPV shell has been cut into pieces fitting in 400 liter drums (Belgian standard waste package).

1. Selection and testing of techniques

1.1. Selection for cutting in air or under water

Based on the experience gained during the previous phases of the pilot project, the BR3 team decided from the early beginning to promote mechanical cutting. For the moment, the band saw is worldwide used and BR3 is proud having been the first one to try this technique remotely and under water.

A first detailed study compared the complete dismantling of the RPV in air or under water (as for the internals in the previous project). For this comparison of dry and wet cutting, the study focused on the following areas :

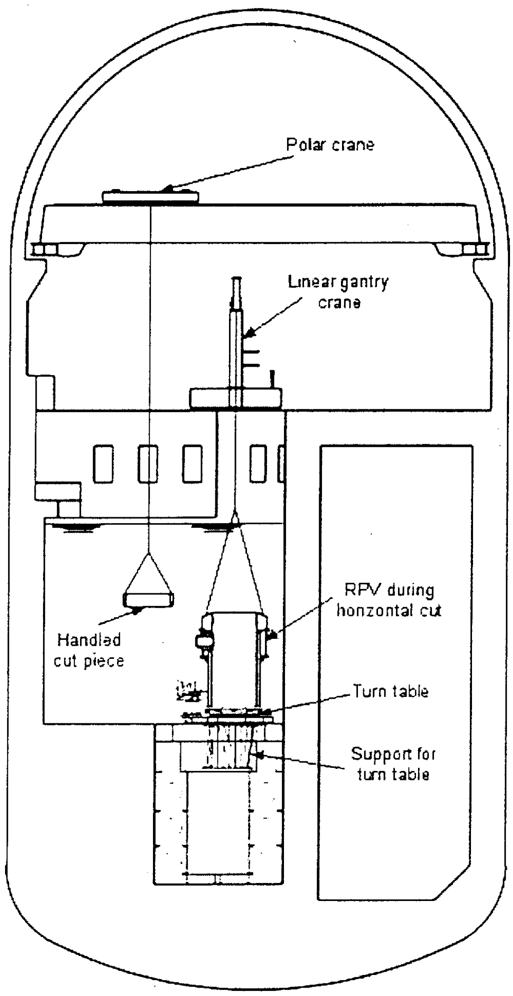
- technical feasibility ;
- the radiation protection and safety of the operators, including the case of equipment failure ;
- the shielding needs to cope with the radioprotection requirements.

As summary, it can be said that the collection and filtration of swarfs when cutting in air represent a difficult problem to solve. The PWR plant configuration with a refuelling pool above the RPV (fit with a purification loop) give an important opportunity to have a good shielding (water).
This media gives also a good accessibility to the whole underwater workshop, built on the bottom of the pool. The RPV being an active part of the refuelling pool, its underwater removal implies modification to guarantee the water tightness of the pool.
The RPV being surrounded by an annular Neutron Shield Tank, the vessel can be submerged (refuelling pool extended to the NST inner wall). Only the three penetrations for the primary pipings have to be closed safely.
So, mainly for technical and safety reasons but also for economical reasons, the underwater dismantling was selected.

A second study was then started to analyse in details two different approaches: the in-situ dismantling where the RPV remains in place while being cut into rings, and the one piece removal, where the vessel is removed in one piece into the refuelling pool, where it will be segmented into pieces ready for packaging.
The advantage of the latter is the accessibility of the RPV and its insulation shroud from the outside, giving the possibility to reuse the dismantling tools and equipment designed for the internals dismantling.
Moreover, this approach simplifies greatly the dismantling of the RPV insulation shroud situated about 100 mm outside the vessel wall.
The basic idea of both studies is to cut the vessel into ring segments and to transport these segments to the storage pool or a special cutting cabin for post-dismantling into smaller pieces.

1.2. Support positioning and driving system

The desolidarisation and the lifting of the RPV give us the possibility to dismantle it in the same way as we did for the internals: using the refuelling pools as underwater workshop.
The figure here after shows the refuelling pool during RPV dismantling. The main support, positioning and driving systems are :



Refuelling pool during RPV dismantling

- ***Support for turntable***

It is a heavy steel construction (design load of 40 t) that is installed on the reactor vessel support ring after removal of the vessel. This support must receive and hold the turntable that is the central dismantling component.

- ***Turntable***

The turntable is made of stainless steel. It allows to clamp the pieces to cut and to present them in front of the cutting equipment. It can be driven manually using long handling tools to drive the gearbox or automatically using hydraulic power on the gearbox.
On the turntable extensions, cutting equipment and/or additional clamping devices can be installed.

- ***Linear gantry crane***

The linear gantry crane of 40 tons was installed above the pool only to lift the RPV out of its cavity, to position it on the turntable and to remove it for service.
During the horizontal cuts, the linear gantry crane always supports the weight of the piece above the cut.

- ***Polar crane***

This crane was used during exploitation of the plant (1 hook 40 t, 1 hook 10 t).
For dismantling purposes it had a new function : tools installation/removal and all other handling operation

such as transfer of cut pieces. It is possible to use the linear gantry crane simultaneously with the polar crane.

- ***Handling of cut pieces***

The cut pieces are rings and, finally, segments. Rings will be handled using a set of three automatic clamping devices hanging at the polar crane. These tools are adapted from the industry to be activated remotely. For the manipulation of segments, a specific tool was designed in order to move and install them on the storage racks.

▲ *Next*

Back

The BR3 dismantling operations and related techniques

Next

Dismantling of the Reactor Pressure Vessel

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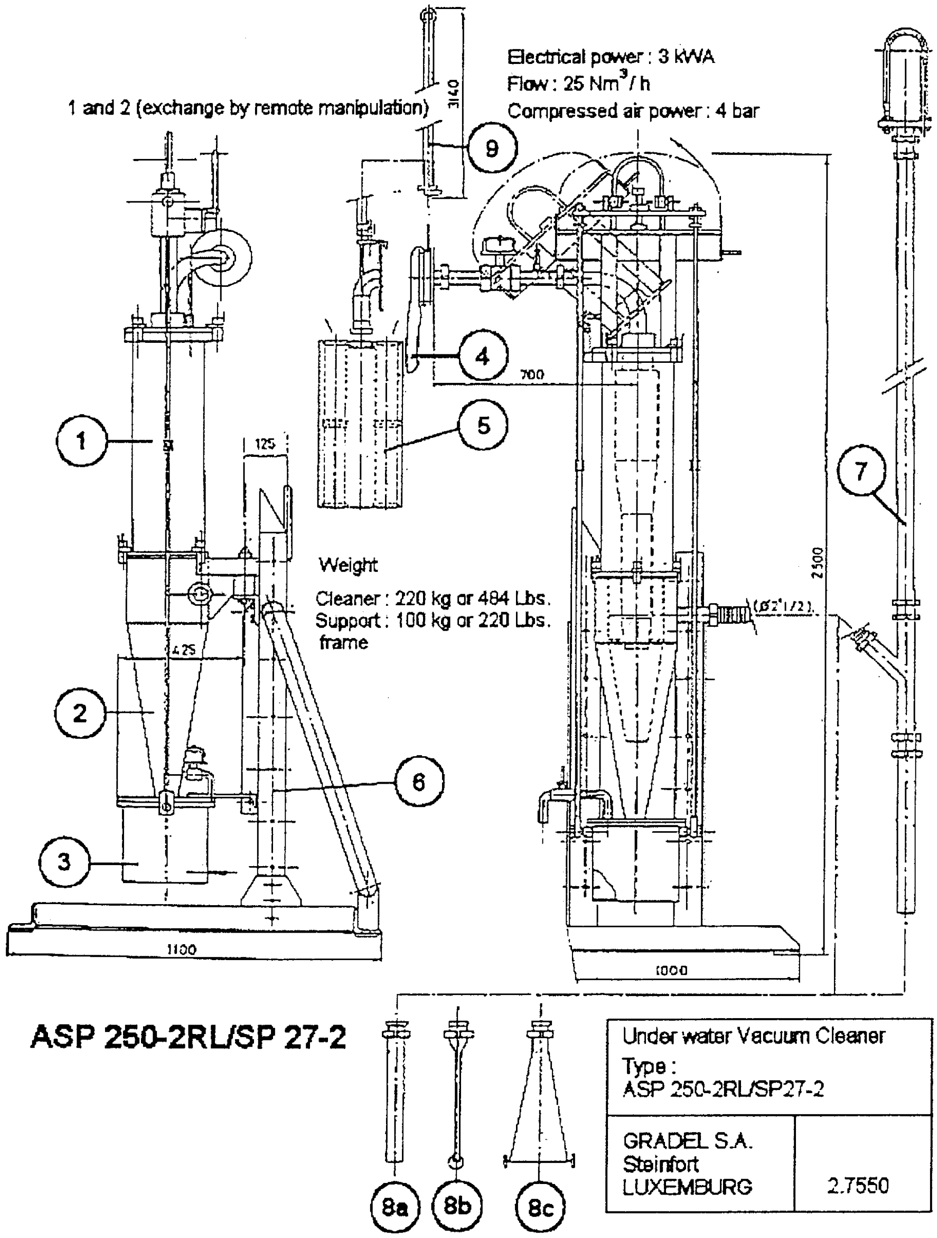
1. Selection and testing of techniques - *(continued)*

1.3. System for the collection and filtration of swarfs and debris

During the dismantling of the RPV, one had to collect two totally different types of (secondary) wastes.
On one side, one had to collect the metal swarfs that were produced with the cutting techniques and, on the other side one had to collect the thermal insulation that surrounded the RPV.
A new bought filtration system collected the swarfs while, for collecting the insulation, a lot of different techniques were applied.

A. The cyclone filtering system

Due to some shortcomings (e.g. multiple swarf blocking, a low filling factor of the waste transport container,...) the increasing wearing (e.g. the sealings) and the high contamination grade of the old filter installation (after 6 years staying under water), the dismantling team decided to buy a new filter installation with better adapted properties.
The new water filtering installation works following the cyclone principle. Here, a conical piece (the cyclone) forces the water into a downward spiralling movement. Due to the radial forces and the gravity, the swarfs ($\rho > 1$ kg/dm^3) are separated and moving downwards.
The main advantages are the lack of pressure drop on the suction side of the pump and the easy collecting of the swarfs at the bottom of the cyclone. The lighter swarfs ($\rho < 1$ kg/dm^3) are trapped in a fine filter situated after the pump.

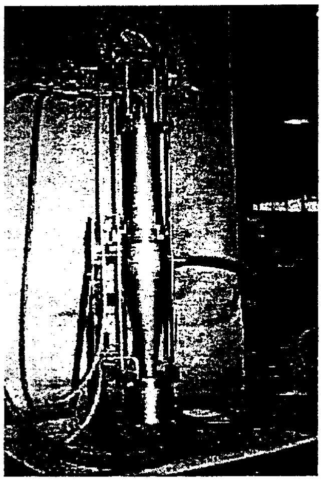


Main subassemblies of the filter installation

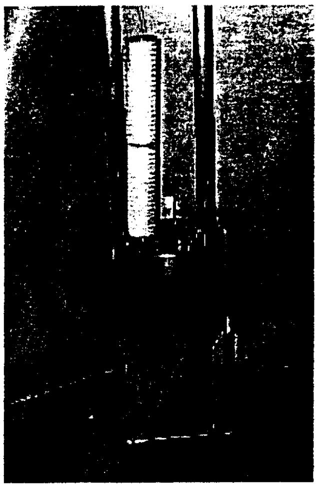
The figure here above shows the main subassemblies of the filter installation.

1	The heart of the installation is the pump with its pump housing.
2	Underneath is situated the cyclone where the separation heavy/light swarfs takes place.
3	A collecting basket under the cyclone gathers the separated swarfs.
4	After the pump, there is a special flange for the connection of a filter bag
5	or a filter housing with five 20" coarse filters.
6	The whole installation rests on a support.
7	A 2½" hose connects a suction device to the filtration installation.
8a, 8b, 8c	This device is provided with one of the three suction mouths.
9	The whole installation is built out of stainless steel and can be used out of or under water. This implies that all the main parts can be removed and handled remotely under water with a special

long handling tool.



General view of the Gradel filtration system



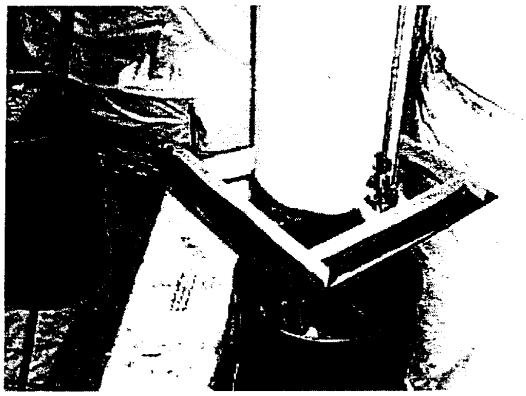
Filtration pot while introducing a cartridge

Filtration pot : one can see the flange allowing the connection of a "*filtration sock*"

▲ The advantages of the new filter installation are :

- The whole installation can be disassembled remotely under water with a long handling tool.
- The maintenance can be carried out quicker and easier. The part that causes a problem can be taken separately out of the water.
- The trapped swarfs do not cause a pressure drop and are collected in a removable collecting basket.
- The collecting basket can easily be taken away and the swarfs can be dropped in a greater basket that increases the filling factor of the transport container.
- The same applies for the fine filters. Because it is possible to take the coarse filters one by one (and not as a whole filter body), they can be put in a special transport rack to increase the filling factor of the transport container.
- The purchase cost of the filter elements reduces: one great collection basket against several strainers and single coarse filters against filter bodies, with repartition plates.

Although the cold tests at the factory were promising, a lot of swarfs plugs were created in the filter system during the horizontal cutting.
They were mostly situated in the conical part of the filter housing. A possible explanation is that the milling process produces long and curled swarfs. In the filtration system, these swarfs have to pass a narrow gap between the conical part of the housing and the protective strainer of the pump.
Due to the specific form of the swarfs, they could easily cling together, forming a plug in the narrow gap. To reduce the number of these plugs, one installed a collecting drum on the suction side of the pump (see figure here below).
This drum is foreseen with a trapdoor at the bottom. This is used for immediate disposal of the swarfs in the waste drums. The trapdoor can be remotely controlled in case of the underwater unloading of high-activated swarfs.



The collecting drum hangs above a waste drum.
An opening system (right from the collecting drum) opens the trapdoor at the bottom to release the swarfs that were collected in the drum.

▲ B. Collecting of the thermal insulation

For the collection of the thermal insulation, not so highly radioactive and heavier than water, one used different techniques.

- *The main filtration system of the pool*
  The heart of this system is a filter containing 64 x 10"-coarse filters of 1 µm and two demineralizers. The filter capacity goes up to 30 m^3/h.
  With this system, the small insulation particles could be trapped. That this system worked well came to expression during the encountered visibility problem caused by this insulation.

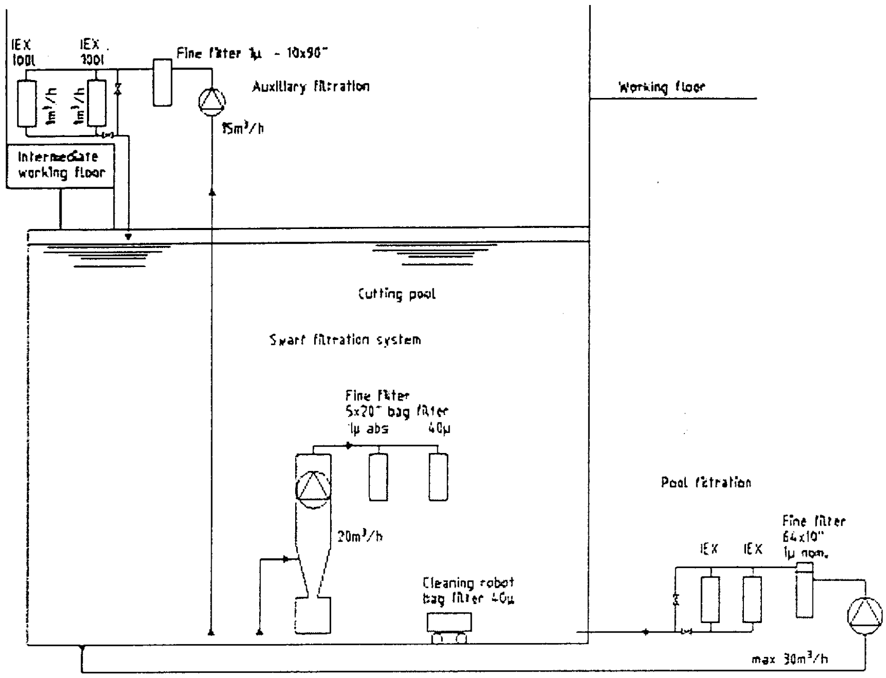
- *External pool filtration system*
  To increase the filter capacity of the pool during the visibility problem, one installed an external filtration system.
  This system has a capacity of 20 m^3/h and has 10 x 30"-coarse filters of 1 µm and at the end there are two demineralizers. One has the capacity of 200 liters, the other has 150 liters.
  The used coarse filters had a fine layer of fibres at their surface, which means that also this system contributes to the collecting of the thermal insulation.

- *Collecting net*
  Before the installation of the RPV on the turntable, one installed a safety net between the RPV and the turntable.
  Along the protection of the moving parts of the table, the net could also collect the sunken insulation. By pulling the net out of the water, a great part of the sunken insulation could be immediately removed out of the water.

- *The pneumatic gripper and plunger pump*
  First, a pneumatic gripper took the remaining big parts of the insulation.
  This technique was not successful. Due to the low density of the fibres they drove away with the water movements caused by the functioning gripper.
  Instead, the operators used a simple plunger pump to collect the largest parts of the sunken insulation. Using the underpressure at the suction side of the pump, they could grip the biggest insulation parts and put the insulation in a collecting basket. Afterwards, the basket came out of the water and the insulation could be dried and disposed of as supercompactable waste.

- *The pool cleaning robot*
  Another selected technique for the collecting of the sunken insulation was the use of a common swimming pool-cleaning robot.
  Cold tests showed out that the robot was capable to collect thermal insulation. The robot is foreseen of a 40 µm-filter bag and can function in a teleoperated mode or in an automatic mode. The robot is mainly used to clean the swimming pool bottoms but it can also climb against the pool walls for cleaning them.
  The robot has done a good job, but nevertheless it has two main disadvantages :
  - The finest filter bag is only 40 µm. This means that only large particles such as long insulation fibres can be collected but not fine particles. On the contrary, the presence of those small particles causes a cloud of "dust" behind the robot that strongly reduces the visibility of the pool water after using the robot for a longer period.
  - Because it is a common cleaning robot, one could not change the filter remotely and/or under water.
    Therefore, one had to be very careful not to collect high or medium activated swarfs.

The next figure shows the different filtration systems available at BR3.



Different filtration systems available at BR3

It may be stated for the interests of decommissioning in generally, that the consequence of dismantling highly activated components with the subsequent removal of radioactivity from the reactor pool should be taken into account in view of water purification problems.
It is recommend to provide for bacteria pollution by treating the water right in time, e.g. with ozone. With this measure the missing effect of the removed radioactivity can be compensated.

Back

The BR3 dismantling operations and related techniques

Dismantling of the Reactor Pressure Vessel

Next

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1. Selection and testing of techniques - *(continued)*

1.4. Full scale testing of techniques

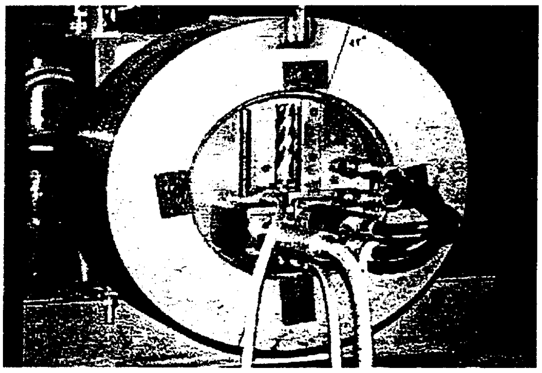
One of the most important lessons learned during former decommissioning experience is that the cold testing of techniques on mock-up can avoid dose uptake and cost due to the unavoidable learning process and the definition of cutting parameters in an easier situation. So all techniques using engineered machines were primarily cold tested.

Five such operations were identified. We refer here to the global decommissioning strategy extensively explained in the following section.
It concerns :

- the internal pipe cutter in order to desolidarize the RPV from the hot and the cold legs near the RPV ;
- the reinstallation of the water tightness of the NST and the reactor pool ;
- the removal of the insulation shroud and its fastening profiles ;
- the horizontal cutting of the RPV using the milling cutter machine ;
- the vertical cutting of the RPV using the band saw machine.

A. Internal pipe cutter

The equipment (see next figure) was tested on a mock-up of the cold leg in a layout as close as possible to the real situation.
The internal pipe cutter is a prototype cutting equipment designed to cut a (very) thick pipe from its inside. The particularity is a very low ratio inside diameter thickness (close to 2.1).
A cylindrical milling cutter is rotating around its axis (~ 300 ... 380 rpm) and can be moved radially inside the pipe and tangentially. So the milling cutter can go through the piping wall following a spiral shape.



Internal thick pipe cutter during cold tests

Rotating speed (cutting speed) : ~ 300 ... 380 rpm
Cutting depth : ~ 15 ... 20 mm
So the radial feed was normally operated in order to keep the cutting depth of the tol at constant values.
Tangential feed speed : ~ 35 mm/min

The cold tests allowed to set up the cutting parameters for this cutting technology which was not exactly the first ones foreseen (the first one foresaw a continuous spiral motion, not an alternate one).

In addition it was possible :

- to optimize the cutting methodology and the cutting parameters ;
- to control the equipment size in the pipe in the neighbourhood of the first elbow of the cold legs ;
- to visualize the operations inside the pipe, to control the cutting data and behaviour of the tool ;
- to study the better management of the chips produced (recuperation & evacuation).

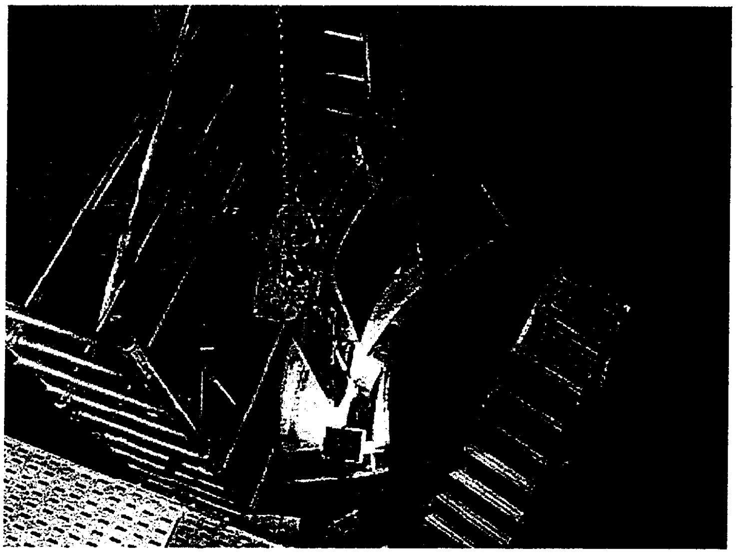
B. Reinstallation of the water tightness of the NST and the reactor pool

The desolidarization of the RPV from the reactor pool and from the primary pipe (by cutting the legs) has destroyed the water tightness of the reactor pool.
Before the lifting of the RPV, the pool must be filled again. So the water tightness must first be reinstalled. The solution found consists in placing and fixing a sealing device on the inner surface of the neutron shield tank in front of each primarily pipe penetration hole.

A first test was organized in order to analyse the remote positioning as well as to control the leak tightness of the system. This first test led to the following conclusions :

- the remote positioning of the sealing piece had to be improved ;
- concerning the leak tightness, irregularities in the construction of the mock-up did not allow to conclude that the system fit perfectly.

So a second test was foreseen on a mock-up with a reviewed design. This test was successful and the actual operation could be prepared (building of the 3 required sealing devices) and scheduled.
The next figure shows the cold test of the introduction of the new sealing device in the adapted NST mock-up.



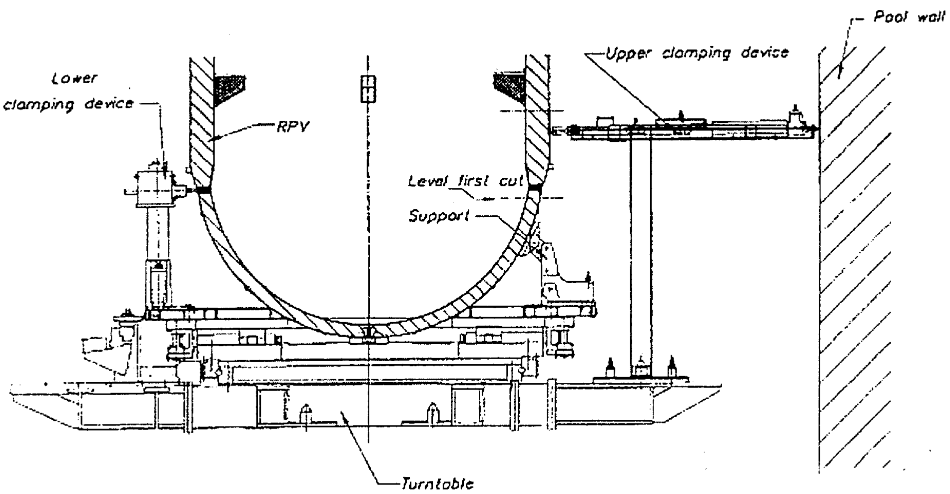
Introduction of the new sealing device in the adapted NST mock-up

C. The removal of the insulation shroud and its fastening profiles

This operation was carried out using a remote hole cutter to destroy the screws maintaining the insulation shell.
The hole cutter is an industrial model that has been adapted to work remotely and under water. A specific supporting equipment and a mock-up were also designed and built.
The full scale test was carried out in a test tank in the non-controlled area of the BR3 plant.

D. The horizontal cutting of the RPV using the milling cutter machine

The chosen dismantling strategy reuses the existing circular saw (and the existing band saw). New tests were necessary to define new machine cutting parameters because compared to the previous phase of the project (i.e. dismantling of the reactor internals), a different sort of base material (carbon steel instead of stainless steel) and another piece thickness (112 mm instead of 25 mm) have to be cut.
Due to the fact that the cutting equipment was contaminated during previous phases of the project, the cold test had to be carried out in the controlled area.
For the circular saw, a new clamping device was designed mainly for the first cut. The purpose is to clamp the upper part of the RPV during and after the cut. The longer part of the RPV is difficult to clamp due to its spherical shape (see next figure).



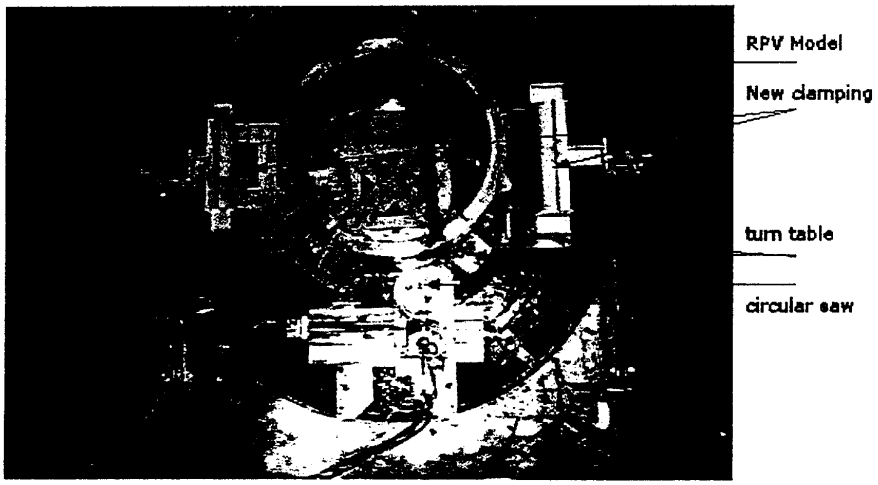
The clamping system for the first horizontal cut.
The cold tests showed that the system was too weak to resist the invloved cutting forces

A mock-up of the reactor pressure vessel was made to carry out tests of the cutting technique with the circular saw.

A first series of tests brought some problems to light :
- the mock-up vibrated a lot;
- the type of sawblade seemed not ideal for the purpose (the metal chips looked more like iron filings with a length smaller than 3 mm).

This *first series* of tests led to :
- the construction of a stiffer clamping system;
- the detailed analysis of the requirements of the cut concerning saw blade design;
- the organization of an extended second series of tests.

The *second series* of tests had as objectives :
- to validate the new clamping system device (see next figure)
- to try and optimize the cutting sequences and their associated parameters.



Mock-up of the RPV clamped on the turntable in the reactor pool using the clamping system

Both objectives were reached and the selected saw blade type had the following characteristics :
- outside diameter : 500 mm

- tooth pitch : 21 mm
- tooth thickness : 6 mm
- coarse toothing : according DIN 1838 (pre cutting teeth about half the thickness but higher by ~ 0.3 mm than the finishing teeth).

Even with an eccentricity of about 0.8 mm (which is much too large), the generated loads during the cutting tests remained at an acceptable level.
The optimized parameters were :

- tangential speed : more than 15 m/min (rotation at about 10 rpm)
- cutting depth : up to 30 mm
- feed speed : ~ 15 ... 20 mm
- chips length : 10 mm as minimum

The eccentricity of the saw blades made that only a quarter of the circumference (and thus of the total amount of teeth) of the saw blades were actually used.
The optimized parameters were used to evaluate the actual job but it was immediately clear that an important improvement could be reached by giving more attention to the eccentricity. A small eccentricity would lead to better parameters and an extended lifetime of the blades as well.

E. The vertical cutting of the RPV using the band saw machine

The foreword mentioned for the milling cutter is also valid for the band saw.

- *Vertical cuts through the nominal thickness of the RPV (112 mm)*
  A few cuts were carried out in the SS cladded carbon steel wall of the RPV mock-up without any problem using a feed speed of about 20 mm/min.
  This immediate success is due to a similar job that the BR3 team already carried out during the previous phase of the project (i.e. the dismantling of the instrumentation collar).
- *Vertical cuts directly through the RPV flange followed by a horizontal cut through vessel-insulation-shroud (all 3 in the same time)*
  The cuts through the flange (thickness 355 mm) were successful at 7 mm/min. Special attention must be paid to the end of the cut of the weld situated below the flange (risk of blade jam).
  The major difficulty for the cutting of the RPV flange is that there is a supporting shroud (thickness ¾") under the flange. This means that for cutting the upper part of the RPV, the band saw has to cut in the same time the flange followed by the vessel + insulation + shroud.
  The influence of the insulation on the cutting performances was unknown and had to be tested. Finally, these cuts were carried out without any problem.

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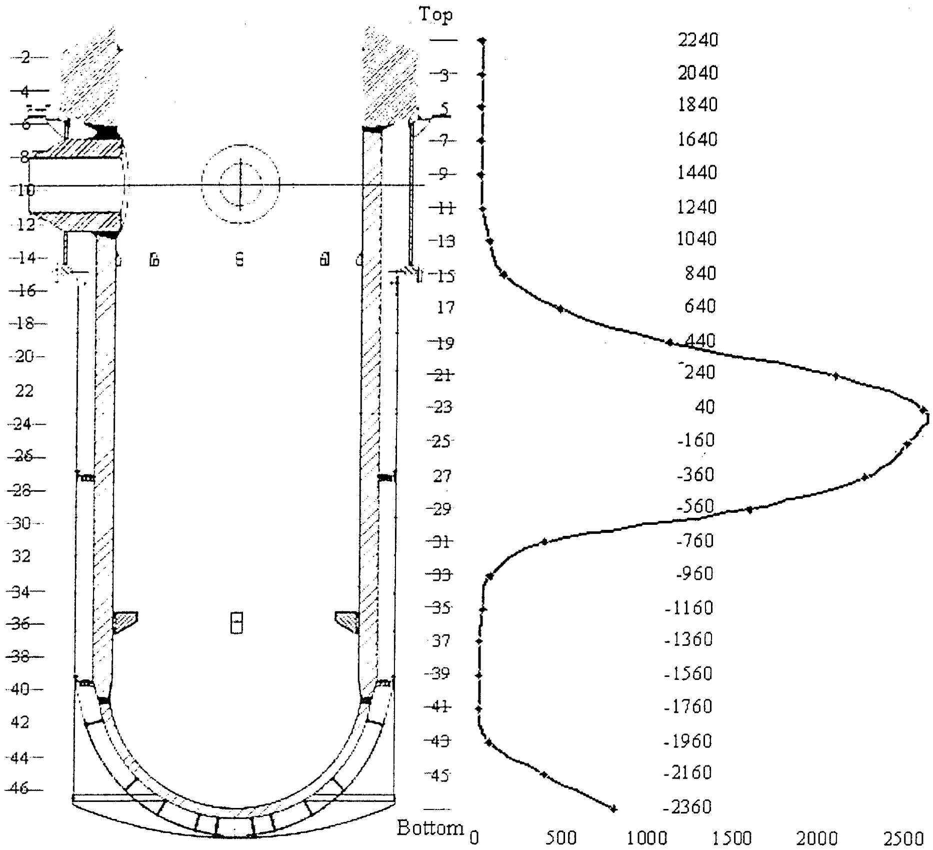
2. Radiological survey and radioprotection optimization

2.1. Radiological characterization of RPV

The figure here below shows underwater contact dose rates measured inside the RPV.
The high dose rate value at the bottom of the RPV is not due to high activity of the RPV material at that position, but to the presence, in the RPV, of secondary waste from the previous phase of the project :

- crud from the primary loop decontamination;
- chips from the in-situ thermal shield segmentation and from the internals dismantling.

mm	Contact Dose rate (mSv/h)				
Distance to the Mid plane	FF-T side	Hot Leg side	Cold Leg 1 side	Cold Leg 2 side	Mean
2240	1,3				1,3
2040	1,3				1,3
1840	2,5				2,5
1640	3				3
1440	5				5
1240	15				15
1040	55				55
840	200	80	70	200	137,5
640	800	200	250	600	462,5
440	1500	800	800	1300	1100
240	2500	1800	1500	2500	2075
40	3500	2200	2200	2500	2600
-160	2500	2500	2500	2500	2500
-360	2000	2500	2500	2000	2250
-560	1000	1800	2500	1000	1575
-760	300	400	500	350	387,5
-960	70				70
-1160	35				35
-1360	9				9
-1560	8				8
-1760	15				15
-1960	70				70
-2160	400				400
-2360	800				800



Contact Dose rate of the BR3 RPV under water (mSv/h)

This fact was confirmed after the post cutting cleaning of the RPV bottom.
In Belgium, 3 categories of radwaste are defined, namely :

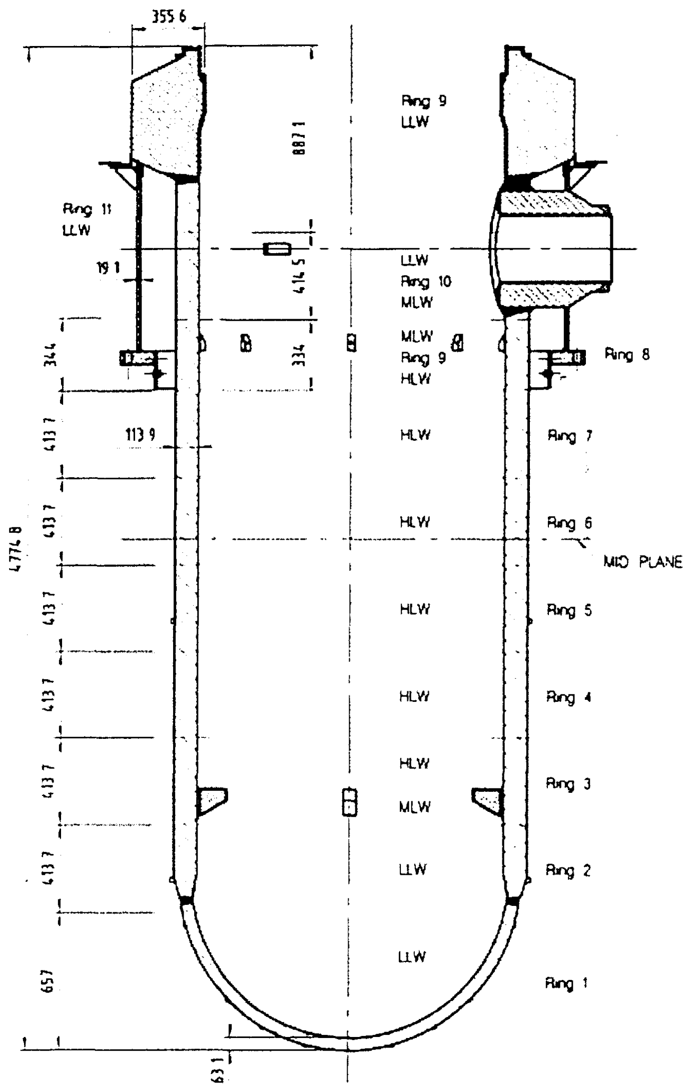
- high level waste (HLW, contact dose rate > 0,2 Sv/h);
- medium level waste (MLW, 0,2 Sv/h> contact dose rate > 2 mSv/h);
- low level waste (LLW, contact dose rate < 2 mSv/h).

As the cost of radwaste conditioning and disposal increases drastically from category to category, it is essential to miminize the amount of radwaste of the higher categories.
In practice, since the RPV will present dose rates (and activities) within the 3 different categories, it is important to fix as exactly as possible the category changing levels. Using samples of the RPV material extracted for other research purposes in 1955, it was possible to calculate the activity at the mid plane level and at the flange weld levels for the main material (CS) and for the SS cladding as well.
The results of the activity measurement (only for 60Co) are shown in table I.

Level	Mean specific activity of Base Metal (CS) (Bq/g)	Mean specific activity of the Cladding (SS) (Bq/g)	Factor SS/CS
Legs	7.78×10^2	9.71×10^3	12.48
Mid-plane	3.56×10^5	7.17×10^6	20.14

Table I : RPV specific activity (^{60}Co)

Both information were introduced to calculate the theoretical dose rate around the final package. This calculation led to the cutting plan showed on the next figure.
The table II indicates thus the wastes categories of the cut rings (expected and realized)



Different levels of the RPV horizontal cut

	Expected waste category	Actual waste category
Ring 1	LLW	LLW
Ring 2	LLW	LLW
Ring 3	MLW	MLW
Ring 4	**HLW**	**MLW**
Ring 5	HLW	HLW
Ring 6	HLW	HLW
Ring 7	HLW	HLW
Ring 8	**LLW**	**MLW**
Ring 9	MLW	MLW
Ring 10	MLW	MLW
Ring 11	LLW	LLW
Ring 12	LLW	LLW

Table II : Comparison of the difference between the expected and the actual waste categories of the cut rings

During the horizontal cutting, chips were collected in order to confirm the specific activity per level. During a cut the milling cutter produces chips in material at various material depths, so it is almost impossible to know where the chips are coming from.
Therefore, this additional measurement only gave indicative value for waste segregation and characterization.
Moreover, sometimes some chips can be coming from the internal cladding of the RPV which is made of stainless steel instead of carbon steel.

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2. Radiological survey and radioprotection optimization - *(continued)*

2.2. Radioprotection optimization to cope with Alara

A. Alara principles

The previous pages described the technical part of the RPV dismantling.
As far as radiological optimization is concerned, the Alara approach has been implemented during the whole RPV dismantling operation.
The Alara principle is the basic idea of the optimization of radiation protection and is based on 3 fundamental principles (ICRP - 26) :

- No practice involving radiation exposure will be undertaken unless its use produces a net benefit (Justification).
- All exposures will be kept As Low As Reasonably Achievable with technological, economic, and social factors considered (Optimization).
- Exposures to individuals will not exceed the limits recommended for the appropriate circumstances (Minimization).

Thanks to our Alara structure, that was set up at the beginning of the dismantling operations of the BR3 reactor, these three principles have been implemented in all stages of the decommissioning.

B. Alara programme elements

The Alara procedure consisted in the following steps :

- An initial detailed subdivision of the work procedure was made.
- 3 work areas were defined with their related average dose rate.
  - At the pool floor (to make some preparation work).
  - At the footbridge above the water level (to perform some manipulations with long handling tools).
  - At the operating deck (to do the maintenance and control the cutting tools).
- A prediction of the required manpower needed for each subtask in the procedure was made.

Using this information, a collective dose and workload prediction was calculated and discussed with the health physics department.
It was also used to verify that all individual doses did not exceed legal limits.
This detailed analysis allowed the tasks giving the main contributions to the total dose to be identified and also the possibilities for reducing the radiation exposure at these critical points to be evaluated.
In order to analyse all data concerning dose and work duration, an electronic computerized dosimetric system is used. Using this data, the actual collective dose could be calculated for every subtask and compared with the predicted values as shown in table III.

▲ *C. Alara practice for the RPV dismantling*

The dismantling of the RPV, was optimized from a radioprotection point of view.
Several options (levels of water, use of shielding material in general, remote controlled tools and long handling tools, need of decontamination, way of decontamination) were evaluated on the basis of a cost-benefit analysis for some of the operations to be performed.

It is obvious that working under water, as for the dismantling of the internals, decreases drastically the dose distributed to the workers (3 m water above highly active components provide a dose reduction factor of about 10^7) and, consequently, that most of the improvements must be done on maintenance operations (where the tools are removed out of the water), on the general waste management (disposal of the segmented pieces in the adequate zone, use of shielding, distance from the workers, ...), on the decontamination ability of the equipment and on the operations duration.

One of the important steps in such an operation is testing the equipment and training the operators in an unrestricted area in conditions similar to the ones they will face in the controlled area.
This was impossible due to the fact that the principal cutting tools, like the bandsaw and the circular saw, already were become "hot" tools while used during the previous dismantling operations. Therefore the "cold" tests on a full size mock-up of the RPV using these cutting tools had to be performed in the controlled area.
The advantage of the RPV dismantling operation was that the operators were used to these tools and had already a lot of experience in working in such circumstances.

Today the RPV dismantling and cutting is finished. The remaining parts of the RPV like the hemispherical bottom, the cover and the insulation shield will be cut in the next dismantling phase using another cutting technique.
In the following table (III) a dose comparison can be found of the different steps in the dismantling process of the RPV.
The data in the fourth column "*First Estimate*" are predictions that were given in the global dismantling procedure (*released in '98*), covering all the different steps foreseen in the dismantling strategy of the RPV.
The column "*Last Estimate*" are the dose calculations as foreseen in the individual procedures (released just before the actual work has started in '99).
The following table gives an overview of the different steps in the dismantling process of the RPV.

Operations	Actual (man-µSv)	Last Estimate (man-µSv)	Delta (%)	First Estimate (man-µSv)	Delta (*Estimation '98*)
Tests in controlled area					
First Series	2 208	4 330	*-49,0*	2 397	*-7,9*
Second Series	2 745	5 000	*-45,1*	2 310	*18,8*
Removal of pool sealing device	2 456	1 320	*86,1*	3 317	*-26,0*
Desolidarisation of the RPV					
NST side	7 855	2 500	*214,2*	6 241	*25,9*
SOD side	2 400	4 680	*-48,7*	2 370	*1,3*
Asbestos removal from core nozzle	3 000	3 285	*-8,7*	2 614	*14,8*
Bolts removal of the RPV	1 400	1 956	*-28,4*	1 564	*-10,5*
Restoring water tightness of the pool					
NST side	14 658	4 245	*245,3*	2 019	*626,0*
SOD side	1 400	1 680	*-16,7*	2 370	*-40,9*
RPV dismantling					
Insulation Removal	5 022	8 340	*-39,8*	6 240	*-19,5*
Horizontal Cutting	4 761	10 080	*-52,8*	6 780	*-29,8*
Vertical Cutting	3 802	1 188	*220,0*	6 189	*-38,6*
Ring 8	443	3 340	*-86,7*		
Ring 11	177	1 032	*-82,8*		
Total	52 327	52 976	*-1,2*	44 411	*17,8*

Table III : Differences between estimations and realized collective doses *(man-µSv)*

▲ The doses shown in the column "*Last Estimate*" are the doses like mentioned on the "*Volet A*".
This "*Volet A*" is a document to be annexed to all dismantling procedures. Every procedure concerning dismantling activities on activated and/or contaminated pieces is discussed with a responsible from the health physics department in order to optimize the collective doses.
The *"Volet A"* has to be completed and approved before releasing the procedure.
The total collective dose differs slighly (- *1,2%*) from the predicted value. Although when looking at the individual procedures, large differences can appear.
When the collective doses were overestimated, most of the time it is due to an overestimation of the workload. In most cases these tasks are more regular or recurring operations.
For the operations with an increased collective dose, the workload was higher mostly due to technical problems or unforeseen circumstances, as is explained hereafter.

- ***Tests in controlled area***

These are the "cold" tests.
They were performed in the reactor pool using the existing (contaminated) cutting tools. While these tests were done in the controlled area, a dose estimation had to be made prior to the execution.
As a result of technical problems during the first series of tests, a second series of tests were necessary.

- ***Desolidarisation of the RPV - NST side***

The realized collective dose was more than 3 times the estimated dose in the procedure due to the following reasons :

- *Preparation*
  - During the preparation of the yard, some modifications were needed to the scaffolds (safety reasons).
  - To reduce the risk of ?-contamination, an internal cleaning of the primary loops with water under high pressure was performed.
  - Due to a possible ?-contamination distribution, a confinement had to be built around every loop to perform this cleaning operation. Initially, these three tasks were not foreseen in the procedure.

- *Actual cutting*

  The tool used to cut those RPV nozzles was specially designed for this task by an external company. Allthough it was tested during a long period in an uncontrolled area, it showed a lot of mechanical and electrical design failures that needed more preparation and maintenance (8 hours instead of 1 hour) for every cut than foreseen in the procedure.
  Therefore the actual cutting time was increased to 480 h instead of 151 h. The mean dose rate is the same (16 µSv/h).

- ***Restoring water thightness of the pool***

The two main reasons for the difference in this operation are :

- Some discrepancies between the "as built" drawings and the reality made the position of these sealing pieces impossible at the time of the first attempt.
  Therefore, the design of the sealing devices had to be revised, and the sealing devices themselves had to be adapted; the placement of these devices had thus to be repeated twice with the corresponding dose uptake.
  The positioning of these devices was finally carried out in June 1999.
- Due to the modifications done on the sealing devices, part of the positioning operations was carried out at the NST level, i.e. close to the RPV in a higher radiation field.

- ▲ ***Concerning the RPV dismantling***

- *Insulation removal*

  For the insulation removal, initially the dose calculation showed a collective dose of 4 380 man-µSv.
  Some weeks after the start of this task, the procedure was reviewed due to several factors. A lot of extra operations were necessary, caused by the fact that the drawings were not "as built". A second factor was the insulation fiber and rust contamination in the water which needed also extra attention.
  Therefore new filtration systems were needed to solve the problem of the visibility of the water.
  All these activities led to a new global dose prediction of 8 340 man-µSv.

- *Horizontal cutting*

  For the horizontal cutting, the dose calculation was done using dose rates measured without any water in the pool.
  Therefore the dose rates were much higher than during the actual cutting work, while this was done under water. This resulted in a dose rate overestimation.
  In combination with a much faster cutting time (cutting speeds were 2 times faster than observed during the cold tests, the gain in collective dose was almost 53 %.

- *Vertical cutting*

  For the vertical cutting the quite important increase in collective dose is due to a lot of unforeseen problems during the under water cutting.
  There were technical problems, problems with water pollution and due to that, the visibility in the water. A lot of time is also lost during the manipulation of the cut pieces to put them in the transportation racks.

- *Ring 8*

  During the cutting of Ring 8, which is the support skirt ring of the RPV, it appeared that this piece was not low active waste like it was characterized before, but medium active waste.
  This had an influence on the final dimensions of this piece. To minimize the waste quantity, it was indispensable to produce smaller pieces. This resulted in some technical problems and therefore it was necessary to review the procedure.
  The best option (radioprotection point of view) was modifying the fixation system on the turntable and cutting Ring 8 under water like the rest of the RPV. However this would be a very time consuming work with an extreme delay in the planning and an extra dose uptake.
  Therefore the health physics department agreed to perform the 8 extra cuts in the dry cutting

workshop using a bandsaw. The contact dose rate was less than 3 mSv/h. The operators were only close to the piece during the clamping on the bandsaw. This also can be seen as an Alara approach, while the dose is kept as reasonably low as possible taking into account the economical factor.

- *Ring 11*

The cutting of Ring 11 is also done in the cutting workshop using a plasma arc torch.
This was already foreseen in the actual procedure. It was acceptable due to the rather low dose rate in contact (> 2 mSv/h) and it could be cut rather fast.
The decrease in collective dose (> 80 % for both operations) can be attributed to an overestimation in workload. Like mentioned before, this is often the case for such tasks with experienced operators.

- ***Conclusion***

Although dismantling tasks are quite specific, compared to maintenance and operational tasks, the Alara programme has been proved to be totally enforceable if such an operation is prepared, followed and analysed from a radiation protection point of view.
The Alara principle has been incorporated into all levels of decision-making which occur during the RPV dismantling operations.

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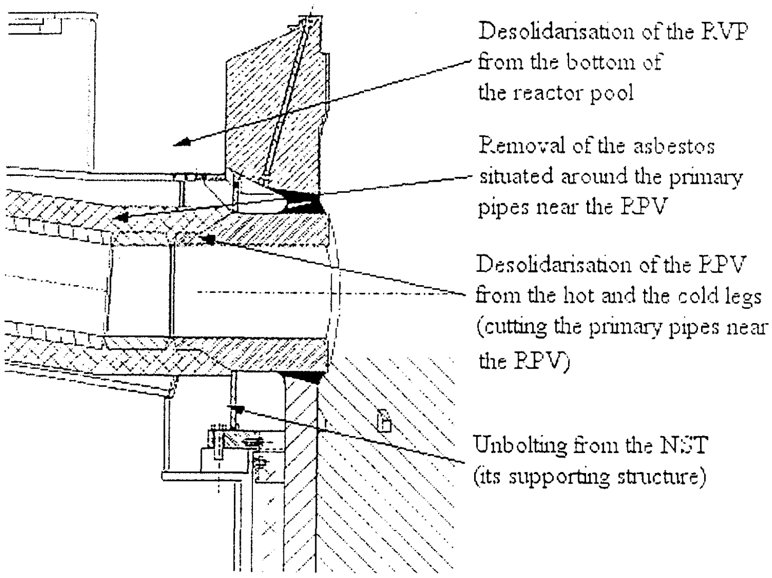
3. Dismantling by various techniques

3.1. Preliminary operations

These operations were executed with a dry refuelling pool, the RPV still being in its cavity, under the bottom of the refuelling pool.
So the access to the pool floor was possible but had to be reduced as much as possible for radioprotection reasons.
The next figure shows a close-up of the RPV connection to the BR3 installation.



Close-up of the RPV connection to the BR3 installation

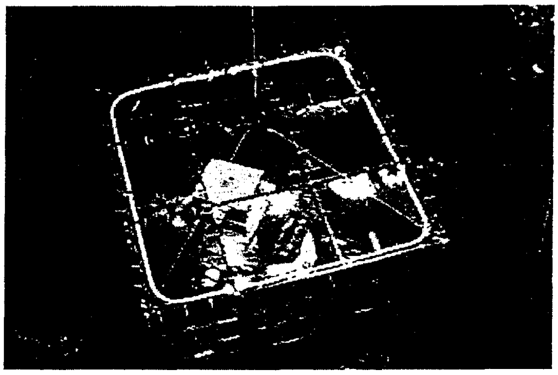
Desolidarization of the RPV from the bottom of the reactor pool

The selected process for cutting the bottom of the reactor pool is the plasma arc torch handled by an operator. The cutting has to be done quickly for limiting the dose uptake of the operations (radioprotection optimization). In addition to this desolidarisation, some cuts at the bottom of the reactor pool were also needed to give access to the fastening bolts of the RPV support flange, to give access to the hot and cold legs thermal insulation and to allow the installation of the sealing equipment for the future water tightness of the pool.

Removal of the asbestos situated around the primary pipes near the RPV

The figure here above and the next one show the removal of the asbestos situated around the primary pipes near the RPV.
SCK•CEN personnel carried out this operation, as the nuclear hazard was estimated to be far above the asbestos hazard.

Nevertheless, to avoid the spread of asbestos fibers, a double confinement was installed in the RPV pool.



First of the double confinement installed in the RPV pool

Desolidarization of the RPV from the hot and the cold legs

Cutting of the primary pipes at the outside of the bioshield (see next figure)

The main operation is the cutting of the pipes at the RPV flange level. Regarding the very tight space available to perform this operation, access was needed through the primary pipes at the bioshield side.
This operation was carried out with a quite common automatic pipe cutter, using two lathe tools diametrically opposed.

Cutting of the primary pipes at the outside of the bioshield

Cutting the primary pipes near the RPV

This operation was delicate due to the fact that access was only available at the inside of the piping.
We thus developed, with an industrial partner, an automatic milling cutter able to cut the necessary thickness.
The challenge was to have a machine fitting into a diameter of 254 mm, able to cup up to 110 mm wall thickness.
Finally, it was decided to make a second cut of the primary pipe connections just above the support flange of the RPV in order to get access to all the RPV fastening bolts.
The cutting tool is an automatic milling cutter with a diameter of 30 mm for the first part of the cut, 25 mm for the second, deepest part.

The intervention procedure was based on the results of the cold test with the internal pipe cutting tool.
The results can be summarized with the following data.

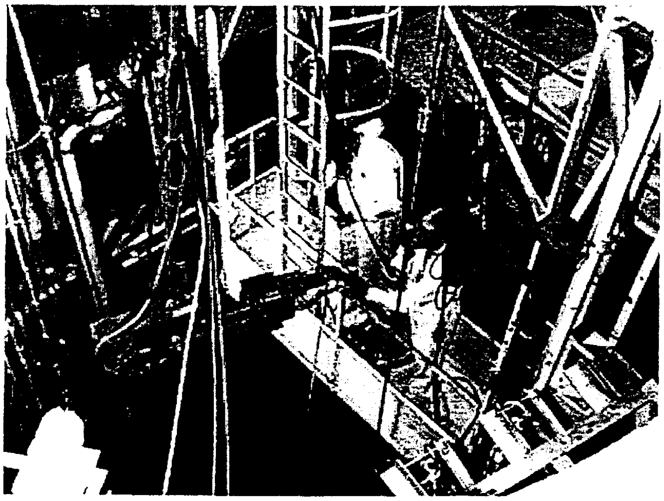
Cut nr	Qty used	Qty foreseen	Item to be cut	Cutter ($Ø_e$ x length) (mm)
1	1	1	Hot leg, thickness ~ 50 mm	25 x 90

2	1	1	Hot leg, thickness ~ 119 mm	25 x 90
	1	1		20 x 125
3	3	2	Cold leg, thickness ~ 40 mm	25 x 90
4	2	2	Cold leg, thickness ~ 108 mm	25 x 90
	3	2		20 x 115

Table IV : Results of the cold test with the internal pipe cutter

- Cutting diagram : the milling cutter went through the piping wall following a spiral shape with a cutting depth of 15 ~ 20 mm, according to the diameter of the tool.
- The quantity of cutting tools required for the intervention was a bit greater than foreseen.
- The rotating speed was taken equal to 300 ~ 380 rpm, according to the diameter of the tool; a reduction of the speed was required while cutting in stainless steel.
- The tangential feed at the tip of the cutter had to be maintained at a constant value of about 35 mm/min.
- The radial feed was such as to keep the cutting depth of the tool at a constant value while following the required spiral shape.

Initially the duration of the intervention was estimated at 10 days / 1 team; in fact the duration was twice as much with 10 days / 2 teams.
This was mainly caused by the weaknesses of the equipment built to fit into the very small diameter of the primary piping system ($Ø_i$ = 250.8 mm for the cold legs) and which had to cut a pipe thickness of more than 100 mm.
As an example after each cut, a control and maintenance of the tool was planned: the foreseen duration of 1 hour for that point was often trespassed (up to 4 to 8 hours). The environment was highly hostile due to alpha contamination in the piping: working close to the tool required always the wearing of overalls and gas masks as a minimum (see next figure).



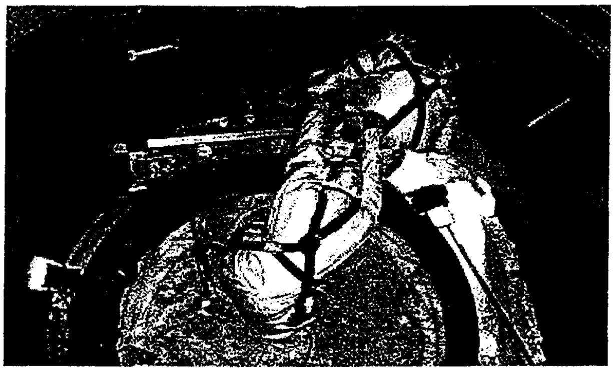
Cutting tool during maintenance (lower level of the mobile footbridge)

Desolidarization of the RPV from the NST

The selected procedure to remove the 24 fastening bolts of the RPV on the Neutron Shield Tank (NST) was the pneumatic unbolting.
Due to the high level of corrosion at the bolts level, this operation took about three times more than foreseen (and thus increased the total dose uptake!).

Reinstallation of the water tightness of the NST and the reactor pool

As the RPV and its primary pipes were part of the pool leak tightness system, we had to seal the openings made by the primary pipe cutting, within is a very tight space.
The operation was carried out with an industrial partner, who developed a system based on an epoxy-based polymer and form-shaped sealing system.
Cold tests were carried out on a real scale mock-up and everything was ready for the installation at the end of March 1999. Unfortunately, some discrepancies between the "as built" drawings and the reality made the positioning of these sealing pieces impossible. Therefore, the design of the sealing devices had to be revised, and the sealing devices themselves had to be adapted.
The positioning of these devices was finally successfully carried out in June 1999 (see next figure).



Positioning of sealing devices

Finally the RPV was ready to be lifted.
A guiding system was also installed, as the mechanical clearance between the RPV and the sealing devices was less than 10 mm. The pressure vessel (22 tons) was then lifted up in one day, using a new gantry crane installed above the RPV pool. The water level in the pool was raised at the same pace as the RPV lifting.
On August 24, 1999 the RPV was removed from its position into the reactor pool for further dismantling.
Direct broadcast was assured thanks to a webcam.

Removal of the insulation shell

The insulation shell was bolted to the RPV through two profiles and on the upper side, it was bolted to the RPV supporting skirt.
It was necessary to remove 60 bolts to free the insulation shell from the RPV. Because of the horizontal position of these bolts, they had to be drilled by a remote hydraulic cutter. For reaching easily the different levels at which the bolts were placed, the remote hydraulic cutter could move up and down along a beam. Here again, cold tests were carried out.
For the actual work, it became almost impossible to localise the screw heads due to a high level of corrosion on the shroud surface. It was then decided to cut the entire circumference of the core shroud using the hole cutter machine. Therefore, it was finally necessary to drill 10 times more holes than foreseen.
An additional problem occurred: some turbidity of the water appeared. This was due to rust but also to the thermal insulation which became breakable into something like dust. Sometimes, the visibility was so bad that the operation had to be stopped. This problem was solved putting into service additional filtration and purification facilities.

Removal of the insulation and the fastening profiles of the insulation shell

The insulation shell was bolted on the RPV by T-shaped fastening profiles and connection pieces on two levels.
Between and on top of these fastening profiles, there was a thermal glass fiber insulation, fastened with a metal mesh. The insulation was also held together with metal strips. On the bottom side of the RPV the insulation was tightened to the RPV with eight strips. The strips were attached on the RPV by bolts throughout the insulation material.
As the mesh was totally rusty, the removal of the insulation was done using a long handling tool. The so liberated insulation fell into a fishing net previously installed on the floor of the pool. By remotely closing the fishing net, the insulation was taken out of the water and evacuated as standard low level waste.
In a first study, it was foreseen to unscrew the bolts of the fastening profiles of the insulation. Finally, the T-shape profiles were remotely attached to the plant container crane and torn up. As the fastening profiles were low activated, their further dismantling was done by hand hold tools.

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3. Dismantling by various techniques -*(continued)*

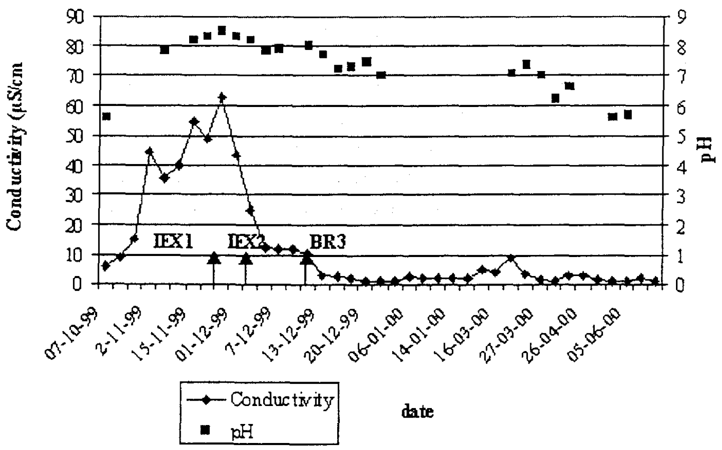
3.2. Turbidity of the pool water

Visibility

During the removal of the reactor shroud and the thermal insulation, a strong pollution of the reactor pool occurred.
The high water turbidity refrained the continuation of the dismantling works. Actions were taken to analyse the origin of the pollution, its nature and to define the corrective actions.

Analysis of the pollution

During the dismantling period from October to December, we observed an increase of the water conductivity from 4 µS/cm (demineralized water) up to 63 µS/cm, the pH increased also from a slightly acidic value of 5.6 up to 8.5.
In the periods of bad visibility, the water was slightly coloured yellow and particles in suspension were measured at a concentration of about 4 mg/l. The activity of the water increased slightly but remained lower than 0.5 Bq/ml in ^{137}Cs and ^{60}Co.
The evolution of the water pollution is shown on the figure here after which gives the water conductivity and the pH observed during the whole period.



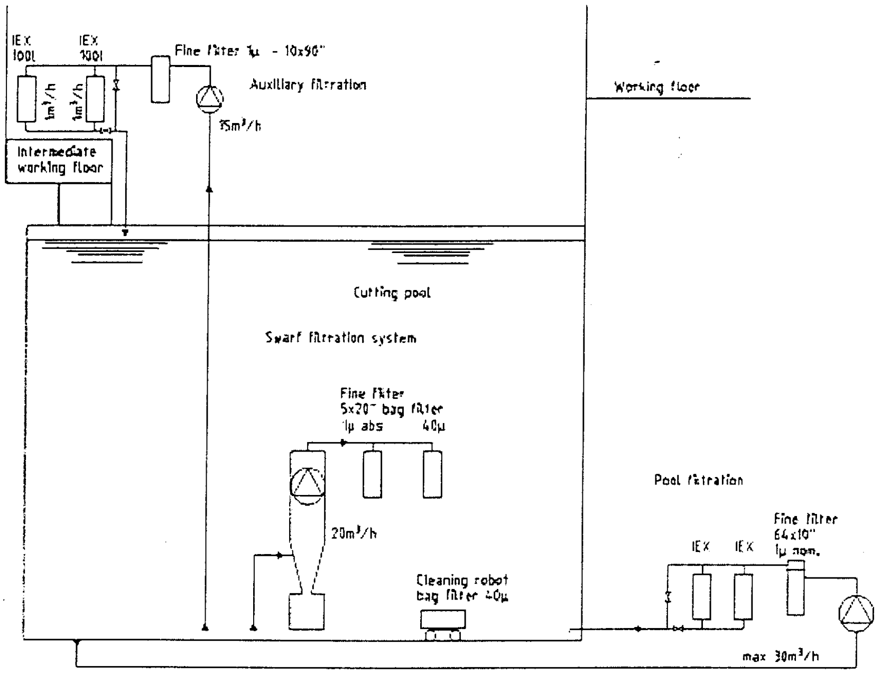
Water conductivity and pH observed during the whole period

During the shroud drilling, the main elements found in solution were B (54 %), Si (29 %) and Na (5 %). The suspended matter was essentially Fe (90 %) and Zn (5 %).
At that time, no significant quantity of insulation fibres was found in suspension.
During the removal of the insulation material, the conductivity was at its highest value (between 45 and 60). The main elements found in solution were B (55 ~ 80 %), Si (8.5 ~ 30 %), Na (1 ~ 10 %), Ca (1.9 ~ 2.3 %) and K (0.3 ~ 1 %). The suspended matter was essentially Fe (64 ~ 95 %), Ca (3 ~ 10 %), Zn (3 ~ 4.7 %), Si (3 ~ 6 %), K (1 ~ 2 %).
The origin of the pollution was at the beginning clearly the release of corrosion products in the water during the shroud drilling: the iron hydroxides rust from the drilling of the carbon steel shroud and the zinc hydroxide from the galvanized wire net used for the fixation of the insulation.
Afterwards, when the insulation material was removed, mineral fibers were found in the water; some binding products, mortar like, used for the fixation of the insulation were also probably released in the water. No

organic pollution or bacterial growth was observed significantly.

Cleaning actions

To recover clear and pure water, it is necessary to remove the particles in suspension and the dissolved ions. Laboratory tests showed that mixed bed ion-exchangers were able to remove the dissolved ions and recover a low conductivity.
Filtration tests were performed at pilot scale using 10 " filter cartridges of different porosity (10, 1.2 and 0.65 µm).
The results showed that it was necessary to use at least a 1.2 µm filter.
The existing purification system of the water of the refuelling pond comprises a filter unit of a capacity of about 20 m^3/h and a 210 liters ion-exchange column of a capacity of 2.5 m^3/h. The filters used are 10 " wound 1 µm polypropylene filter cartridges (64 filters). The ion exchanger is a homogeneous mixture of a strongly acidic cation resin and a strongly basic anion resin.



Different filtration systems available at BR3

This system is insufficient to deal with the strong pollution observed; moreover it appeared that the ion exchange column was saturated. Therefore, several actions were undertaken :

- installation of an additional mobile filtration unit of a capacity of 20 m^3/h pumping directly in the pool;
- replacement of the saturated resins;
- installation at the outlet of the mobile filtration unit of two mobile ion exchange columns of a capacity of 2 to 3 m^3/h.

All these systems were progressively put into operation so that the situation could finally be controlled around mid-December.
Afterwards, the visibility could be kept under control except during short periods of peak pollution corresponding to release or resuspension of rust or fibres. To maintain a high water quality, it was necessary to replace regularly the saturated filters or the resins.
During the whole cutting period of the reactor pressure vessel, the production of secondary waste amounted to about 2.9 m^3 of burnable cartridges (25 charges) and 0.7 m^3 of burnable ion-exchangers.

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3. Dismantling by various techniques - *(continued)*

3.3. Horizontal cutting of the reactor pressure vessel - Circular saw

On the basis of the tests some improvements were made :
- the use of sawblades with a much smaller eccentricity allowed to increase the feed speed as well as the life span by a factor of about 4;
- a semi-automatic system for the recuperation and evacuation of the metal chips avoided complete jam of the circular saw during operation; moreover it was necessary to place a cyclone before the inlet of the suction pump as a protection against a flow of too much chips at a time.

The following figure gives a sight view of the reactor pool while cutting the RPV with the circular saw.

Sight of the reactor pool with the RPV, the saw on its XY table, the metal chips evacuation system with the cyclone and the already cut rings on their storage structure

The results of the intervention can be summarised as follows (see also table here under):
- Cutting diagram : a complete horizontal cut of the vessel was achieved in two steps :
  - linear cutting of introduction sectors with the translation of the circular saw working on a fixed vessel;
  - circumferential cutting of the remaining material with rotation of the vessel with a fixed circular saw.
- As the life span of the sawblades improved significantly, the quantity of cutting tools required for the intervention was much smaller than foreseen: 9 blades made according to DIN 1838C with an external diameter of 500 mm and a tooth pitch of 19.63 mm.
- The rotational speed of the sawblade was maintained constant around 10 rpm.

Initially the duration of the intervention was estimated at 110 shifts on the basis of a mean feed speed of 15 mm/min; in fact the overall duration of the intervention was reduced by 34 shifts (leading to a total duration of 76 shifts), mainly because of much better performance of the saw blades (feed speed up to 80 mm/min and longer life span) as expected after the cold tests.

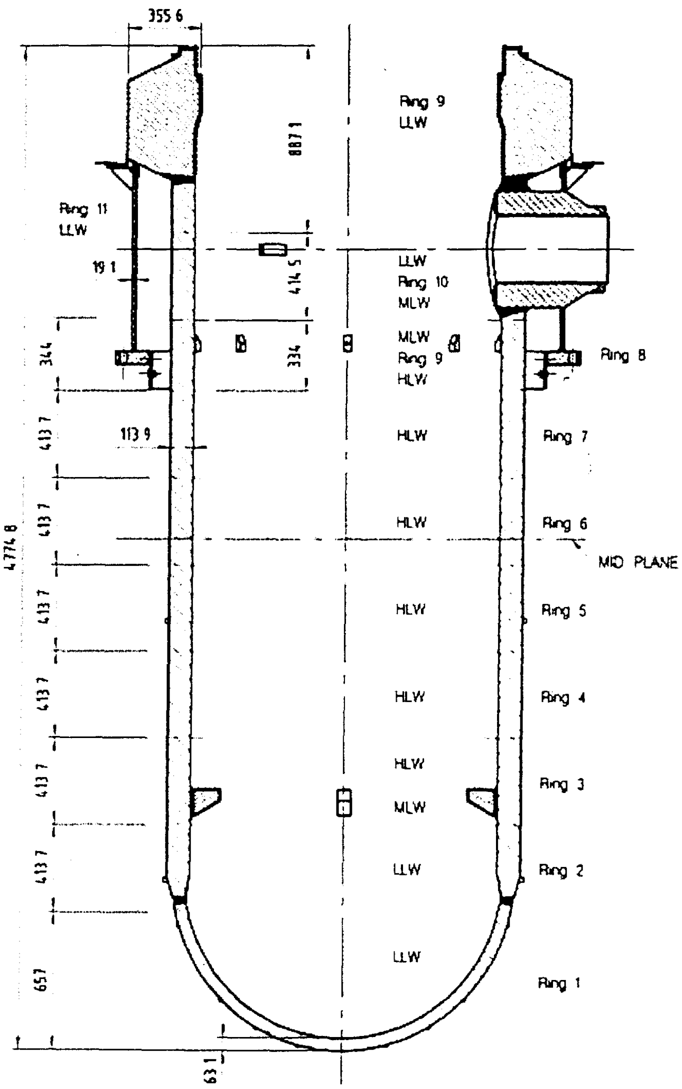
Steps	Predicted Shifts	Predicted Workload (man-hour)	Predicted Dose (man-mSv)	Actual Shifts	Actual Workload (man-hour)	Actual Dose (man-mSv)
Preparation	3	72	0.444	7.0	149	0.432
Ring 1	11.5	372	1.788	10.0	409	0.709
Ring 2	13.5	324	1.108	8.0	222	0.393
Ring 3	12.5	300	1.028	4.5	116	0.293
Ring 4	11.5	276	0.876	4.5	116	0.185
Ring 5	12.5	300	1.028	5.5	147	0.332
Ring 6	12	288	0.952	4.5	119	0.224
Ring 7	12.5	300	1.028	5.5	137	0.258
Ring 8	5	120	0.54	2.5	45	0.099
Ring 9	12.5	300	0.992	6.0	134	0.385
Total	106.5	2 652	9.784	58.0	1 594	3.31

Steps	Cut surface (dm^2)	Average cutting speed (mm/min)	Cutting capacity (mm^2/min)	Swarfs (kg)	Number of blades
Preparation	-	-	-	-	-
Ring 1	7.9	22.0	89.4	3.8	2
Ring 2	14.2	28.5	175.5	6.8	1
Ring 3	14.2	36.7	222.6	6.8	1
Ring 4	14.2	31.7	202.6	6.8	1
Ring 5	14.2	25.5	177.9	6.8	1
Ring 6	14.2	28.7	222.9	6.8	1
Ring 7	14.2	27.4	136.0	6.8	1
Ring 8	3.1	41.2	198.1	1.5	(1)
Ring 9	14.2	21.0	-	6.8	1
Total	110.4			53.0	9

Table V : RPV dismantling - Horizontal cuts

3.4. Vertical cutting of the RPV - Band saw

The following considers only the rings that were cut with the bandsaw.
These rings are showed on the figure here after.

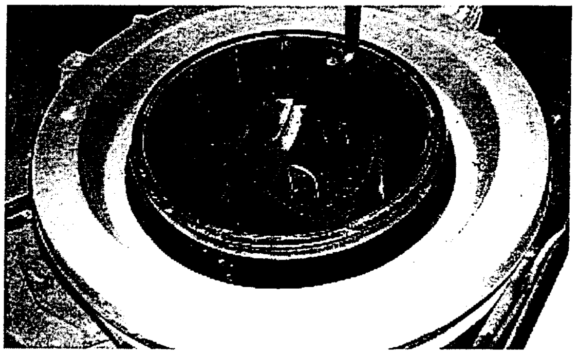


Different levels of the RPV horizontal cut

Ring 1 (bottom of the vessel) and also the vessel head will be cut with another technique later on in the dismantling project.
Ring 11 (low activated) was cut with the plasma torch in our dry cutting workshop.
Ring 8 (also foreseen as low activated) with a reciprocating saw (hands on techniques).

- *Cutting of the vessel flange*

After the last horizontal cut, the first piece presented to the bandsaw was the reactor flange (rings 12 and 10; see figure above).
First 15 vertical cuts of about 950 mm length were performed in the vessel flange in order to make "teeth". The last cut was done above the hot leg in order to turn the sawblade from a vertical cutting position into a horizontal cutting position at a well-defined height. Then a hydraulic motor drove the turntable so that the bandsaw could make a horizontal cut and cut off the "teeth", one by one.
The cut pieces were put immediately in standard 400 liters-drums and could then be removed as low activated waste (see figure here below).



Standard 400 liters drum loaded with a flange piece

The remaining ring (ring 10) could then be cut in the same way like the other rings.
The vertical "teeth" cuts were quite difficult for two reasons. First due to the thickness of the flange (up to 360 mm) and second due to the presence of a big weld and the sometimes complex cross section.
This is reflected in the high consuming rate of the blades (17 against 3 for a normal ring - see table VI).

Ring	Predicted Shifts	Predicted Workload (man-hour)	Predicted Dose (man-mSv)	Actual Shifts	Actual Workload (man-hour)	Actual Dose (man-mSv)
Preparation	-	-	-	3	55	0.124
12 (vert) 12 (hor)	9.5	163.25	0.368	32.5	669	1.219
10	2.5	43.5	0.127	13	254	0.435
9	2.5	42.75	0.124	6	147	0.207
7	2	34.8	0.095	9	196	0.345
6	2	34.8	0.095	7	160	0.356
5	2	34.8	0.095	4.25	65	0.135
4	2	34.8	0.095	5.25	116	0.360
3	2	34.8	0.095	5.5	75	0.222
2	2	34.8	0.095	5.5	134	0.404
Finishing	-	-	-	1.5	33	0.049
Total	26.5	458.3	1.189	96.0	1 905	3.856

Ring	Average cutting speed (mm/min)	Cut surface (dm^2)	Cutting capacity (cm^2/min)	Swarfs (kg)	Number of blades
Preparation			-		
12 (vert)	4.7	424.0	29.61	66.14	17
12 (hor)	22	56.57	10.42	8.80	
10	10.4	50.1	11.8	6.85	6
9	11.0	53.1	13.0	7.27	3
7	8.0	56.0	9.0	7.66	3
6	9.4	55.5	10.7	7.60	3
5	9.0	55.5	10.3	7.60	3
4	7.8	55.5	8.9	7.60	3
3	10.6	55.5	12.1	7.60	2
2	8.9	55.5	10.2	7.60	3
Finishing			-		
Total	110.4			53.0	9

Remarks

- With the average cutting speed is meant the average speed of the saw blade when it goes through the base metal.
- With "preparation" and "finishing" is meant the start and termination of the cutting yard.
- Only the rings cut with the bandsaw were taken into account.

Table VI : RPV dismantling - Vertical cuts

- *Cutting of the rings*

The cutting of the remaining cylindrical rings of the RPV shell showed no problem.
However, the used cutting speed was much lower than during the cold testing (9.5 mm/min instead of 19 mm/min).
This can be explained as follows: for the two first rings, the operating team used a lot of sawblades. In order to avoid the same high consumption of blades during the next rings, the operating team started the next cuts with a reduced feed compared to the feed used during the cold tests.
This went well so that the team would not take any risks and remained at the same cutting speed.

- *Summary*

The table VI gives the most important data concerning the vertical cutting.
One remarks the great underestimation of the workload and the collective dose.
The causes were :

1. The unforeseen visibility problems due to the presence of the thermal insulation (the same problem already encountered during the horizontal cutting).
2. The lower cutting speed that was used (50 % lower) and the fact that the blade had to be changed more often than foreseen, doubled the cutting time for the ring.
3. But the main cause was a totally wrong estimation of the waste manipulation after the cutting of the ring, i.e. the filling of the waste racks, and certainly when there were problems with the weak designed guiding pins (see further: encountered problems).

3.5. Encountered problems during the cutting of the RPV

Although the bandsaw technique proved its reliability during previous dismantling phases and the cold tests did not came up with any major problem, we encountered few problems from which some lessons had to be learnt.

- *Visibility problem*

After the removal of the insulation shell and the thermal insulation, a part of the thermal insulation remains between ring 10 and ring 11 that caused serious water turbidity problem during the horizontal cut by the bandsaw.
While the blade cut off the "teeth" in the vessel flange, it had also to cut through the packed thermal insulation.
Due to the moving of the blade, a small part of this insulation went into the pool water decreasing the pool visibility as a result. This visibility became very poor after the horizontal cut with the movement of the remaining rings on the turntable (rings 11 and 10).
A cleaning of the turntable and a filtering of the pool water was necessary. As a matter of fact, we had the same problem as with the removal of the thermal insulation.
The removal of this thermal insulation and the filtering of the water took 13.5 shifts.

- *Mechanical bandsaw problems*

In the preparation phase of the vessel dismantling, we revised the bandsaw.
The changing of all the bearings and seals and also a global check up of the bandsaw were part of this maintenance. Nevertheless, a mechanical problem occurred in the beginning of the vertical cutting campaign. The drive wheel of the band saw machine is equipped with an inner gear bolted to the main wheel port. Some of these bolts broke at the beginning of the campaign. In a first attempt, one replaced these bolts by others of a higher quality, but the problem remained.
After a close examination of the driving mechanism, the shaft of the hydraulic motor showed too much clearance, probably caused by a damaged bearing, giving too many vibrations to the main wheel. Changing of this bearing solved the problem.
In total, the solving of this problem took 4 shifts.

- *Waste racks problems*

A last problem concerned the waste racks where the cut pieces were put after the cutting of the ring.
The guiding pins of the waste racks (the cut pieces had to be put between these pins) were too fragile. This means that when such a pin was hit during the placement of the pieces, it bend.
This caused a lot of problems with the further placing of the remaining pieces in the waste rack and there was even a great risk that the hitting piece fell out of its gripping tool. This sort of problems caused a great delay in the cutting operations of the ring.
This problem is difficult to quantify but one example can pinpoint the problem: the waste manipulation of ring 10 took 5 shifts while normally this manipulation should ask less than 2 shifts.

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4. Waste production

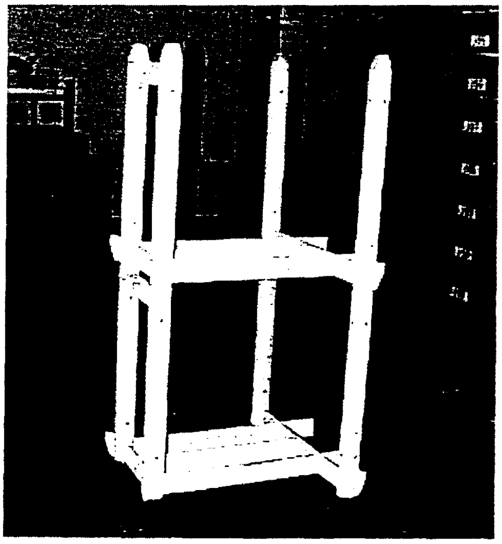
During the vertical cutting a certain amount of waste was produced.
This waste is normally split up in two groups, the primary waste and the secondary waste.
One considered as *primary waste* the cut pieces of the vessel, and as *secondary waste* all the waste produced while using this technique. In this case the swarfs and the filters of the different pool filtration systems are considered as secondary waste.

4.1. Waste removal system

For the removal of the high and medium activated waste of the reactor internals dismantling (former operation), the team designed a removal system based on two racks where the cut pieces were stored and immobilized and a support structure on which the two racks glided.
This system fulfilled the different Belgian waste conditioning and safety requirements.
Using this system, some shortcomings appeared :

- The weight of the racks and supporting structure was not optimised (as the total weight per drum is limited by the national radwaste agency).
- The manipulation of this type of racks was difficult.
- The gliding of the racks over the support structure was not always easy due to an asymmetrical position of the cut pieces in the waste rack.

For the dismantling of the RPV, the team redesigned an improved system as shown on the following figure.



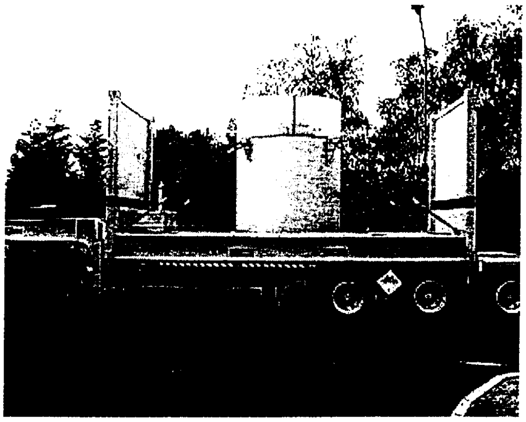
The prototype of the renewed rackdesign.
The racks can be bolted together and are manipulated with only one lifting device

The layout of the waste rack itself remained, but a bolting system replaced the support structure.
Four bolts were placed in the tubes of the upper rack and a threatened hole was put in the tubes of the lower rack. A special long handling tool was designed to bolt the two racks together.
Beside that, the team carried out a renewed mechanical resistance study in order to minimize the weight of the waste rack.

The improved design of the waste removal system had thus 2 main advantages :

- less weight than the old system and so a higher effective waste load;
- just one, already existing lifting device was necessary to carry out the different rack manipulations.

This waste removal system with the cut pieces was then put in the waste transportation container (see next figure) for further waste conditioning and intermediate storage at Belgoprocess.



View of the waste transport container for the high level radioactive waste

4.2. Primary waste

The dismantling of the RPV produced solid waste in the *three Belgian waste categories* : high, medium and low activated metal waste (or HLW, MLW, LLW).
The table below represents the quantity of the produced waste types.

Waste type	Quantity (tons)
HLW	5.15
MLW	7.22
LLW	10.5
Total	22.42

Remark : The vessel head and bottom and the insulation shell are not included in the table above. They will be cut with another technique in the next phase of the dismantling project.

Table VII : Quantity of produced waste types

4.3. Secondary waste

The quantity of swarfs forms a first group in this type of waste.
The calculated weight of produced swarfs was 386 kg. The amount of swarfs was also distributed over the three waste categories.
The table below compares the calculated weight with the measured weight of swarfs.

Waste type	Calculated (kg)	Measured (kg)
HLW	170	86.9
MLW	80	-
LLW	136	-
Total	386	-

Remark : The medium and low activated swarfs were distributed over different 200 l-drums (2 drums MLW and 5 drums LLW) of which the weight was not measured.

Table VIII : Difference between calculated and measured weight

The HLW-swarfs are collected in special designed waste baskets which are compatible with the renewed waste rack design.
The great difference between the estimated mass of high activated swarfs and the removed mass to the waste conditioner can be explained: first of all, one ring was estimated as high activated, but in reality this ring was medium activated. This means that the estimation was already too high.
Secondly, as the difference in density between the swarfs and the base metal can be as high as 8 (1 g/cm^3

swarf density; 8 g/cm^3 base metal density), the specific activity of the swarfs can be reduced by the same factor.
Therefore, the swarfs originating from metal being near the lower limit of a category can fall into the lower category only by this density difference.
The high activated waste will be transported to Belgoprocess, the Belgian waste conditioner, with 9 transports (primary waste and high activated swarfs) representing 9 400 liters drums or a waste volume of 3.6 mm^3.
The filters of the different pool purification installations form a second main group.
First we have the original and main pool filtration unit where the filter consists of 64 coarse filters of 10 ".
Second there is the auxiliary pool filtration equipment were the filter consists of 5 coarse filters of 30 " and two ion exchangers.
Table IX below gives the volumes of these filters for the different main steps as well as the volume on worn tools for both techniques.

	Main pool filtration	Auxiliary pool filtration (coarse)	Auxiliary pool filtration (ion exchangers)	Tools
	(dm^3)	(dm^3)	(dm^3)	(dm^3)
Horizontal cutting	184	12.15	100 *	8.8
Vertical cutting	184	36.45	100 *	11.8
Total	736	73.2	-	20.6

* The same columns were used for horizontal and vertical cutting so that no difference could be made between the two.

Table IX : Volumes of filters and worn out tools

▲ 5. Comparison between both cutting techniques

Two different mechanical techniques were used to dismantle the same piece. Therefore it is interesting to make a comparison between both techniques.
To have a relevant comparison, only the cutting of similar rings was taken in to account. This means in practice that the values gained during the cutting of the reactor flange are not taken into account in the comparison.
Also the workload and the received collective dose related to the visibility problem during the use of the bandsaw, are not calculated in the compared data.
To compare both techniques, all the data of the circular saw are recalculated to a cut section equal to that of the bandsaw (436.7 dm^2). This means multiplying the circular saw data with a factor 3.96.
This recalculation gives us the opportunity to calculate the ratio circular saw (CS) and bandsaw (BS) that is given in the 6th column of table X.
The table gives also the ratio calculated after the internals dismantling to show the influence of the piece thickness (vessel: 112 mm compared to the 25 mm of the internals).

	Unit	Circular Saw (gross values)	Bandsaw	Circular saw equivalent	Ratio C.S. eq /BS	Previous ratio (reactor internals)
Cut section	(dm^2)	110.4	436.7	436.7	1	1
Total cutting time (8h shift)	(Days)	58	63.5	230	3.6	0.99
Total workload	(man-hours)	1 594	1 236	6 312	5.1	1.05
Observed collective dose	(man-mSv)	3.31	2.64	13.11	4.9	1.02
Kerf width	(mm)	6	2	6	3	3
Removed metal volume	(dm^3)	6.62	8.73	26.21	3	3
Number of used blades		9	26	36	1.4	0.49
Blade cost per unit	(Bef)				4.9	26.9
Total blade cost	(Bef)				6.8	13

Table X : Comparison of the data of both techniques

The table tells us :
- The removed metal volume (secondary waste) is three times higher with the circular saw.
- The circular saw is a machine with a high blade cost.
- The circular saw is more labour intensive when the thickness of the piece increases.

This is shown very clearly by comparison with the ratio of the internals.

Indeed, cutting a thick walled piece asked much more time. First of all we had to execute several cutting passes with the circular saw and some introductionary cuts for avoiding a vertical drift of the blade.

Conclusions :
- The bandsaw is more preferable than the circular saw when the thickness of the piece increases, and when the bandsaw can be used.
- The bandsaw produces less secondary waste (swarfs and worn blades) due to a lower kerf width.
- The use of circular saw requires much more stiffness of the workpiece (to avoid vibrations) than the bandsaw.
- When the both techniques are applicable, the bandsaw should be preferred but, however, the choice between both techniques will be mostly be guided by the geometry of the work piece and the accessibility of the environment.

6. CONCLUSIONS AND MAIN RESULTS

The complete remote dismantling of the BR3-PWR pressure vessel proved the actual feasibility of such an operation.

The results showed that :

- quite simple mechanical tooling (i.e. band saw, circular saw, ...) adapted for working remotely and under water can be used for such operation;
- important time was spent on the preparation and pre-segmentation operation;
- external parameters and non-conformity between drawings and reality led to the main difficulties.

Nevertheless, the main operation was carried out safely, in an economical way and keeping the objective of waste volume minimization.
This project completes a process started with the dismantling of the reactor internals, and which has led to compare, test and use different remote dismantling techniques for segmenting the most radioactive part of a nuclear power plant.
Although the BR3 plant is a low rated plant, the pressure vessel geometry (thickness) and material can be of interest for commercial size power plants.

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Dismantling of highly radioactive reactor internals

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1. Description of the reactor internals

The BR3 reactor was first equipped with original Westinghouse internals, from the reactor designer and supplier.
In 1964, after 2 cycles, for an experiment called "Vulcain", the internals were exchanged (except the thermal shield) and the old, original ones, were stored in a shielded chamber situated in a corner of the refuelling pool. It was first foreseen to reload the original internals when the Vulcain experiment was achieved, but this was never done, and the Vulcain internals remained in the reactor until the final shutdown. These Vulcain internals have thus undergone the last 9 operating campaigns of the reactor.
The whole project covered the dismantling of all the reactor internals :

- the thermal shield (phase 1) ,
- the "Vulcain" internals (phase 2) ,
- and the original "Westinghouse" internals (phase 2_{bis}).

1.1 The thermal shield and the "Vulcain" internals

The following figure (figure) shows a general view of the reactor vessel with the "Vulcain" internals. Through a short cylindrical support skirt, the vessel rests on the Neutron Shield Tank, a large annular carbon steel vessel surrounding the reactor vessel and filled with chromated water, which was maintained at low temperature (50° C) during reactor operation. The inner diameter of the reactor vessel is about 1.5 m.
The core (fuel active length about 1 m) is surrounded by the thermal shield, a stainless steel cylinder of 76 mm wall thickness. The thermal shield is the heaviest of the reactor vessel internal pieces (weight about 6 t) and the only one never removed from the reactor vessel during the active plant life.
The other internal pieces can be classified into three sub-assemblies :

- 1. The lower core support assembly (LCSA)
  It is a stainless steel assembly of about 4.5 m height and 4 t weight. The outer diameter ranges from 1.2 to 1.5 m and the maximum wall thickness is 40 mm. This assembly rests on a cylindrical shoulder at the upper part of the reactor vessel.

- 2. The upper core support assembly (UCSA)
  It is a stainless steel piece, loaded above the reactor core, maintaining the fuel assemblies in place and, through a spray box, providing part of the safety injection water flow directly above the reactor core. The outer diameter of this piece is about 1 m and its weight is about 300 kg.

- 3. The collar and its associated instrumentation basket.
  The reactor vessel collar is a thick ring of carbon steel lined inside with stainless steel (wall thickness : 194 mm; outer diameter : 1715 mm ; height : 310 mm). The collar, which was not part of the original design of the reactor vessel, was inserted between the vessel flange and the vessel cover in order to give access to the vessel interior for pipes and instrumentation.
  Two sets of O-rings and bolts of extended length are used to fasten the collar and the cover on the reactor vessel. The collar is representative (thickness and material) of PWR vessels.

The neutron-induced activity of the internals surrounding the reactor core (LCSA at core level and thermal shield) (see Table I) is very high. The activation of the reactor collar is very low due to its longer distance from the reactor core. Moreover, there were also guide tubes for the control rods and associated gripping mechanism, as well as the control rods themselves, constituted of borated stainless steel tubes.

1. Thermal shield (in October 1991, i.e. 4.25 years after shutdown)

Level vs. midplane (mm)	Depth (overall thickness = 76 mm) (mm)	Specific activity (GBq/kg)	Specific activity at shutdown, based only on ^{60}Co (GBq/kg)
- 238	0 - 19	56	98 46
	19 - 38	n.d.	34
	38 - 57	26.3	
	57 - 76	19.7	
+ 248	0 - 19	n.d.	70
	14 - 38	40	36
	38 - 57	20.4	34
	57 - 76	19.3	
+ 735	0 - 19	n.d.	10.3
	19 - 38	5.9	2.8
	38 - 57	1.6	6.8
	57 - 76	3.9	

2. Lower Core Support Assembly (in June 1994, i.e. 7 years after shutdown)

Level vs. midplane (mm)	Cut number	Specific activity (after 7 years) (GBq/kg)	Specific activity at shutdown, based only on ^{60}Co (GBq/kg)
- 633	1	3.4	8.5
- 189	2	67.0	168.2
- 213	2_{bis} (core baffle)	125.0	313.9
+ 311	3	19.0	47.7
+ 263	3_{bis} (core baffle)	28.0	70.3
+ 801	4	1.2	3.0
+ 1111	5	0.015	0.038
+ 1918	6	0.0019	0.0048
+ 1626	7	0.00092	0.0023

Table I : Specific activity of the dismantled internals

Above : Thermal shield (in October 1991, i.e. 4.25 years after shutdown)

Under : Lower Core Support Assembly (in June 1994, i.e. 7 years after shutdown)

1.2 The Westinghouse internals

The Westinghouse internals are all made in stainless steel (AISI 304). They consist of two main subassemblies : the upper core support assembly (UCSA) and the lower core support assembly (LCSA). They also include a few other subassemblies which are described hereafter.

- *a. Guide tubes hold down plate and ring*
  These two pieces are 25.4 mm thick plates. The guide tubes hold down plate is embedded inside the guide tubes hold down ring to form a subassembly of the reactor. The overall diameter is about 1450 mm.

- *b. Guide tubes*
  There are twelve guide tubes. A guide tube has a cylindrical geometry with a conical top end (diameter = 171.5/154 mm ; H = 1657 mm). These tubes were used to guide the control rod into the reactor core.

- *c. Guide tubes support plate*
  The guide tubes support plate is a 32 mm thick plate with twelve big holes for the guide tubes.

- *d. Upper Core Support Assembly*
  It comprises two main subassemblies : the upper core support barrel and the upper core support plate. The upper core support barrel is a cylindrical assembly with top and bottom flanges. The cylinder comprises a circular opening for the water flow to the hot leg. The upper core support plate is a rigid assembly of two perforated plates joined by welding to a spacer ring at their circumference. The overall height of the upper core support plate is 101.6 mm. The upper plate of the assembly supports twelve dashpot stops.

- *e. Lower Core Support Assembly*
  It comprises the following subassemblies :
    - the lower core support barrel ;
    - the reactor core barrel ;
    - the reactor core baffle ;

- the lower core support plate ;
- twelve control rod extension shrouds ;
- a tie plate at the end of those shroud tubes.

The lower core support barrel consists of a conical section and cylindrical section with top and bottom flanges. The cylindrical section has a nozzle (hot leg) and two diameter guide spacers. It is bolted at its bottom flange to the core barrel and the core baffle.

The core barrel is a cylindrical piece (diameter = 1188/1130 mm ; H = 1693 mm) with top and bottom rings. It contains the core baffle (bolted to its upper ring). At its bottom, it is fastened to the lower core support plate by 18 bolts placed top down.

The reactor core baffle consists of a square structure made of plates and ribs with circular top and bottom flanges. Its main thickness is about 6.35 mm.

The lower core support plate is similar to the upper core support plate. The upper part of the lower core support plate supports guide blocks. The control rod extension shrouds are twelve cylindrical pieces (H = 1286 mm, diameter = 168/154 mm) hanging at the lower core support plate. They are bolted to the lower plate of the core support plate by cap screws. The tie plate is attached by cap screws to the lower end of the shroud tubes.
After 30 years of cooling down, the activity of the internals was still quite high and presented a contact dose rate at the core level much too high to allow direct operation without shielding.
The following table gives a summary of the activity of the pieces and a measure of the remaining crud contamination activity still present on the internals.

Level vs. midplane (mm)	Cut number	Specific activity (after 31 years) (GBq/kg)	Specific activity at unloading based on ^{60}Co (GBq/kg)
- 1300	Control rod extension shrouds	0.008	0.47
- 403	W1 (Reactor core barrel)	1.805	106.31
- 785	W2 (Lower core support plate)	1.206	71.03
- 413	W3 (Reactor core baffle)	3.454	203.43
+ 47	W4 (Reactor core barrel)	1.998	117.68
+ 37	W5 (Reactor core baffle)	8.114	477.89
+ 497	W6 (Reactor core barrel)	0.565	33.28
+ 487	W7 (Reactor core baffle)	2.014	118.62
+ 997	W8 (Lower core support barrel)	0.009	0.53
+ 929	W9 (Upper core support plate)	0.073	4.30
+ 1373	W10 (Lower core support barrel)	0.001	0.06
+ 1402	W11 (Upper core support barrel)	0.001	0.06

Table I : Specific activity measurement of the Westinghouse internals

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2. Dismantling of the thermal shield and comparison of three main cutting techniques - *(plasma arc torch, EDM, mechanical cutting)*

The second part of the phase 1 contract concerned the selection and comparison of cutting techniques fo dismantling highly radioactive internals, followed by the testing on a first internal : the reactor thermal shield The thermal shield is a thick stainless steel cylinder (thickness : 76.2 mm or 3 inches, height : 2.43 m, externa diameter : 1.397 m) which surrounds the core and was never unloaded during the whole life of the plant.

Three different cutting techniques were selected for the dismantling of the thermal shield : the plasma ar torch cutting, the electric discharge machining (EDM or sparkling erosion) and the mechanical cutting using a milling cutter.
This allowed to compare three different cutting methods belonging to three types of techniques : a thermal one (the plasma arc torch), an electric one (the EDM) and a mechanical one (the milling cutter). The comparison concerned the amount of generated *secondary waste*, the cutting duration, the operator's dose uptake and the easiness of the operation. Other cutting techniques, like laser cutting or high pressure abrasive water jet cutting, were also considered but finally not selected because they were still under development and not yet completely *"mature"* for the type of cutting, the environment (under water and radiologically aggressive) and the thickness of the material.

The philosophy was to use existing and proven technologies and to adapt them to the environment and to the application. Great care was paid to the dose forecast (ALARA approach), the secondary waste and the operator's safety. Considering the secondary waste production foreseen, it was decided to perform plasma cutting in a closed chamber, located in the reactor refuelling pool, allowing to circulate and filter the water as well as the air situated above the water level (filtration of aerosols, evacuation of the produced hydrogen). The foreseen waste packaging technique implied to cut the thermal shield into pieces having as main dimensions 500 mm x 540 mm x thickness. Regarding the geometry of the thermal shield itself, it was necessary to solve the problem of the spacer pins situated at the top level. Indeed, these pins were screwed in place after the loading of the thermal shield in the pressure vessel and then fastened by welding. The pressure vesse presenting diameter restrictions above the thermal shield, it was impossible to unload the complete piece as such, or even a ring of it. Therefore, it was decided to cut vertically the first ring in situ.

The complete cutting responds to the different constraints given for the operation :

- packaging dimensions < 500 mm x 540 mm ;
- presence of the spacer pins ;
- comparison of different cutting techniques ;
- no thermal load on the RPV wall ;
- plasma torch cutting performed in closed chamber.

After the design and procurement phases, cold testing of the three techniques were carried out on full-scale mock-ups.

These cold tests allowed to determine the best cutting parameters, to train the operators and to solve some *"youth illness"* of the installations. These cold tests or trials proved to be very efficient in predicting the cutting parameters and in helping to forecast the dose uptake and to optimize the radiation protection.
Moreover, the only part which was not fully tested during the cold tests, for practical reasons, was also the one which gave afterwards the most important problems to solve once in the controlled area. This is one of the *first important results* of this phase of the project : *"for important operations or activities related to highly radioactive pieces, the use of cold trials on full-scale mock-ups is really necessary and cannot be avoided".*

It saves finally money and time by allowing to solve problems in an easy environment prior to enter the nuclear environment. The thermal shield was cut into 40 pieces in 4 months. The pieces were stored at the bottom o the refuelling pool of the plant, awaiting their transfer to the deactivation pool using a specifically designed container.

Table III hereafter gives the main results of the cutting campaign.

Cutting method	Cut length	Kerf width / depth N° x Length	Cutting speed (1)	Time for cutting + prepa-ration	Effective cutting speed	Metal removed from thermal shield	2^{nd}ary waste volume	Ratio waste vol/cut length (1)	Dose uptake (6)	Dose uptake per cut length
	[m]	[mm]	[mm/min]	[hours]	[mm/min]	[dm³]	[dm³]	[dm³/m]	[mSv]	[mSv/m]
Vertical EDM	2.92	7/76.2 8 x 365	0.6	184	0.27	1.622	96 (2)	33	7.5	2.56
Horizontal Mechanical	4.15	4/18-20 4 x 4150	6	64	1.08	1.262	26.7 (3)	6.4		
Horizontal Mechanical	4.15	4/18-20 4 x 4150	6	80	0.88	1.262	26.7 (3)	6.4	8.3	0.66
Horizontal Mechanical	4.15	4/18-20 4 x 4150	6	48	1.44	1.262	26.7 (3)	6.4		
Horizontal EDM	4.15	7/76.2 12 x 346	0.6	240	0.29	2.208	128 (2)	31	7.1	1.71
Vertical Plasma	14.98	11/76.2 24 x 482 8 x 426	300	136	1.83	12.55	459 (4) (5)	31	9.2	0.61

(1) For 76.2 mm wall thickness. For EDM, this corresponds to an equivalent speed, the way of cutting being different from the two other methods

(2) Fine filters containing activated EDM particles

(3) Coarse filters containing activated chips

(4) Coarse, intermediate & fine filters; demineralizer columns not completely saturated

(5) Not taking into account the waste produced by the air filtration system

(6) The preparatory phase, including the unloading of the reactor internals gave rise to a total dose uptake of 22.6 man-mSv

Table III : Thermal Shield Segmentation

The mechanical cutting method produced the smallest secondary waste volume while taking finally less than twice the operating time of the plasma arc torch cutting. Taking into account the long preparation and cold test time as well as the remediation of the plasma dross collecting system problem, the overall operation time is almost similar.
For the dose uptake, the very slow operation speed of the EDM implies a long staying time in the controlled area and in the ambient dose rate (even if low), and then a quite high dose uptake.
The next table gives a summary of the preceding results :

Parameter	Cutting speed (through SS, 76.2 mm thick)		Average effective cutting speed		Dose uptake	Secondary waste volume (for same cut)
	[mm/min]		[mm/min]			
Cutting method	Absolute	Relative	Absolute	Relative	Only relative values	Only relative values
EDM	0.6	*1/10*	0.28	*1/4*	*~ 3*	*~ 5*
Mechanical	6	*1*	1.13	*1*	*1*	*1*
Plasma	300	*50*	1.83	*1.6*	*~ 1*	*~ 5*

Table IV : Summary of the Thermal Shield dismantling results

For the produced waste, the values in the table did not take into account the used pool filters of the plant filtration unit. The particles and dust dispersed by the cut in the reactor pool were mostly trapped in these filters.
The distribution of these secondary waste among the different cutting techniques has been made in the following table, using the filter exchange dates and the cutting campaign period as reference.
This is indeed approximative, as some particles can have settled on the bottom of the pool. Nevertheless, it gives also an indication of the pollution of the pool induced by the cutting process.

	Number of cartridges	Net Waste volume [dm^3]
EDM	3	138.2
Mechanical	1	46.1
Plasma	7	322.6

Table V : Use of the pool filtration cartridges from the plant installation (one filter exchange is constituted by 64 cartridges of 10" long)

The results presented above led finally to give the preference to mechanical cutting for the next phases of the project. Indeed, the main advantages of the mechanical cutting can be summarized as follows :

- well known technique, used in workshops; must only be adapted to work under water ;
- secondary waste type (chips or swarfs) easily trapped, with a good filling factor of the filter ;
- low amount of waste if the tool and the kerf are thin ;
- no emission of smoke, gas and dissolved ions ;
- overall operation duration comparable to other cutting processes.

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3. Comparison of different mechanical cutting techniques for dismantling highly radioactive reactor internals

Thanks to the experience gained during phase 1, described above, and taking into account the general geometry of the highly radioactive internals (all internal pieces to be cut have a general shape of revolution of elementary surfaces), it was decided to apply mostly the mechanical cutting technique for their dismantling, where possible.
Two main techniques were selected : the circular saw and the band saw in association with a so-called turntable. The goal was to cut the highly active internals in segments which have a size fitting closely to the final disposal waste package (400 l waste drum).
The main results of the operations using both techniques are summarized hereafter.

3.1 Circular sawing

The circular saw was used during phases 2 and 2_{bis} of the project and was foreseen to cut the long cylindrical workpieces horizontally. It was fixed on an extension of the turntable on which the workpiece was fastened. The circular saw support has 2 degrees of freedom : the X-axis for the feed of the blade into the workpiece (available stroke : about 1 m) and the Y-axis giving the cutting depth (available stroke : about 320 mm). All the movements of the sawingmachine as well as the rotational speed of the blade were controlled and monitored remotely.
The following table gives the different blades used for this underwater segmentation.

Saw blade		Number of cutter	Rotational speed	Peripheral speed	Cutting depth/pass	Theoretical maximum cutting depth
Diameter	Pitch					
[mm]	[mm]		[Rpm]	[m/mim]	[mm]	[mm]
400	19	3	10	12.6	25 to 30	100
480	20	3	8	12.6	25 to 30	100
610	6	2	7	13.4	25	205
610	19	2	7	13.4	25	205
660	20	2	6	12.4	25	230

Table VI : Number of saw blades used

The first two showed the best results and straight cuts. But for deep cutting or long distances between the cutting head and the workpiece, the blades with larger diameter (Ø 610 mm or Ø 660 mm) had to be used. The blade with a Ø 610 mm with a pitch of 6 mm was mostly used for cutting thin material (i.e. 1.6 mm thickness) of the baffle.
The following table gives the main results obtained during the cutting operations for the two sets of internals.

	Cut section	Cutting time during swarfs production	Total cutting time	Average feed during swarfs production	Average production for the whole cut
	[dm²]	[min]	[min]	[dm²/min]	[dm²/min]
Vulcain internals	94.55	6024	23040	0.025	0.004

Westinghouse internals	63.24	3295	15006	0.020	0.004
Total / Mean	157.79	9319	38046	0.023	0.004

Table VII : Circular saw cutting results

The total cutting time includes the operations for changing the clamping pieces, for cutting and for the maintenance of the sawing machine.
During the swarfs production the average feed is about 0.023 dm²/min. In fact, the feed fluctuates from less than 0.004 dm²/min to 0.055 dm²/min following the shape of the workpiece to be cut.

3.2 Band sawing

The band saw was foreseen to carry out the vertical cutting of the LCSA and of the reactor vessel collar as well as the segmentation of the plates.
The band saw has a throat of 500 mm and a vertical cutting capability of 960 mm. These dimensions allowed to cut the different pieces to dismantle using the same machine. Thin material (like the core baffle, thickness 1.6 mm) as well as thick annular pieces like the reactor vessel collar (carbon steel clad with stainless steel, overall thickness about 200 mm) can be cut by the machine using adapted saw blades and cutting parameters. Different blade types (2/3 Teeth per inch, 4/6 Tpi, 6/10 Tpi and 10/14 Tpi) were used depending on the type of cut to be carried out. Moreover, the machine is able to make horizontal as well as vertical cuts (e.g. for plate segmentation), the blade guides being able to rotate around a horizontal axis.
For vertical cutting, the feed is achieved by a controlled vertical motion of the band saw frame while, for horizontal cutting, the workpiece itself is rotated, using the turntable.
The following table gives the main results obtained during the cutting campaigns.

	Cutting section	Cutting time during swarfs production	Total cutting time	Average feed during swarfs production	Average production for the whole cut
	[dm²]	[min]	[min]	[dm²/min]	]dm²/min]
Vulcain internals	216.65	5269	52320	0.041	0.004
Westinghouse internals	160.60	4112	21546	0.09	0.007
Total / Mean	368.14	9381	73866	0.040	0.005

Table VIII : Band saw cutting results

The total cutting time includes the operations for changing the clamping pieces, for cutting and for the maintenance of the sawing machine.
The feed during swarfs production varies quite sensibly, from 0.007 (thin thickness : 1.65 mm) to 0.105 dm²/min (thick annular piece, overall thickness : about 200 mm).

3.3 Swarfs collection

It was originally planned to collect the swarfs during the cut by means of a suction frame surrounding the saw blade. Swarfs were also collected in a funnel with a collecting basket placed under and inside the workpiece. At the end of the horizontal cut of the Vulcain internals, due to frequent blocking of the suction system, the swarfs were not collected anymore during the cut, but were pushed into the funnel, by a water jet after each cut. The remaining swarfs located on the turntable were then sucked off at the end of each cutting campaign by using a straight suction hose.
The total calculated weight of produced swarfs for the whole cutting campaign was 133 kg from which 104 kg were collected by the two methods described above. The remaining 29 kg were located at the bottom of the pool and in the reactor pressure vessel and were removed by suction afterwards.

3.4 Summary and conclusion

The next table gives a summary of the two cutting campaigns on the "Vulcain" and "Westinghouse" internals. This gives an overview of the man-power and the dose uptake for the performance.

	Cutting length	Cut section	Man-power	Dose uptake	Working time
	[m]	[dm²]	[man*hours]	[man*mSv]	[hour]
Vulcain internals	104.12	311.2	3815	26.98	1256
Westinghouse internals	89.02	223.84	2278	9.25	609
Total	193.14	535.04	6093	36.23	1865

Table IX : Summary of the man-power and dose uptake during the present cutting campaign

The next table gives the qualitative differences and performances of both techniques as they were applied in BR3 for this project.

	BR3 Circular saw	BR3 Band saw
Cutting force	~ 7 500 Nm	~ 800 N
Overall volume of the machine	small	large
Horizontal cutting position (level)	only one	several
Cut type	horizontal	horizontal & vertical
Shape of the workpiece	cylindrical or linear without too complex cross section	free with complex cross section
Kerf width	6 mm	2 mm
Depth of passes	25 mm depth per pass	only one pass limited by the opening of the saw blade (500 mm)
Maximum actually cut thickness	192 mm	365 mm
Maximum possible cut thickness	~ 230 mm (for Ø 660 mm)	~ 500 mm (for the BR3 band saw)
Feed for the same cut cross section in SS Thickness : 25 mm	5.8 cm²/min	6.02 cm²/min (*)
190 mm	3.05 cm²/min	7.9 cm²/min
Swarfs dimensions	great	small
Maximum height of the workpiece	independant	limited (here 900 mm) (by the return way of the blade)
Free space around the workpiece	without requirements inside or behind the workpiece, only limited by the stroke of the saw (320 mm)	both sides of the workpiece free with a minimum of Ø 700 mm free behind the workpiece for the wheels

* Limited by the maximum linear feed rate of the machine (25 mm/min).

Table X : Qualitative difference between the band saw and the circular saw

Both techniques showed to be reliable, usable and efficient.
In the BR3 project, both techniques were complementary. It is obvious that the circular saw technique produces more volume of secondary waste (metal swarfs) due to a greater kerf width. The removed volume of metal (swarfs) is three times higher than with the band saw.
The average cut production (overall cutting speed) for the whole cut was 1.25 times higher with the band saw. Thus, where it is possible (depending on the height, the shape and the existing access on both sides of the piece), it is better to work with the band saw.
In other words, during the general design of a cutting campaign of reactor internals, it is important to minimize the circular saw work and to maximize the band saw within the limits of their capacities, but both remain complementary.

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4. Other techniques used for the dismantling of the reactor internals

The dismantling of the reactor internals was based on some main techniques and machines as described above. A lot of auxiliary techniques were needed to carry out some specific tasks. These techniques were used to prepare the internals before cutting, to execute some dismounting, to complete a cut begun with a main technique or as a back-up technique. These auxiliary techniques are presented hereafter with the main results obtained and the lessons drawn.

4.1 Hydraulic shears

The hydraulic shears allow to cut, very fast and without any production of chips, pieces presenting a relatively small cross section. Remote work at a distance of up to 7 m was possible, the shears being fixed at the end of a long handling tool. If direct vision is not possible, the positioning can be checked using an underwater television system. The cut capacity reached with the hydraulic shears was the following :

- in full metal, about 30 x 7 mm ;
- for tubes OD/ID = 42/35 mm.

The replacement of the cutters was needed when the cut edge was not sharp enough anymore. The frequency of replacement depends on the type of cuts which are carried out and cannot be planned easily. Some sets of spare blades are always needed.
The hydraulic shears were used on almost all the BR3 internals :

- on the Vulcain LCSA to dismantle the lower end of the internal and to cut auxiliary tubes ;
- on the Vulcain UCSA box to cut the six columns, joining the upper part of the internal to the lower part ;
- on the Vulcain instrumentation basket which was almost made of plates and tubes.

The hydraulic shears were also used for two other applications.
First, with the band saw machine, to cut the blade and then free the machine when a blade was blocked in the kerf. After placement of a new blade, a new cut could be started just near the first one.
The second application concerns mostly manipulation of pieces up to about 20 kg. The shears can be used as a gripper holding pieces which are dismounted.
The negative feature of the hydraulic shears is the place required around the piece to cut (about 10 cm) for opening of the jaws. Moreover the body of the used machine is about 60 cm long and its diameter about 15 cm.

4.2 Core drilling

The core drilling machine is a mechanical cutting technique which allows to "drill" holes of diameters up to 50 mm. The one used is pneumatically driven and the feed motion is given manually with a long handling rod. The secondary waste produced is composed of chips and a cylindrical core which is extracted from the annular cutter.
The use of this tool was required to avoid problems on the band saw machine during horizontal cuts. Indeed, the saw blade can deviate from its perfect horizontal way, the blade then becoming overloaded and can finally

block and/or break. If the saw blade meets at some place a hole, it can go back to its original cut level and the cut can be continued without problems.
On the Upper Core Support Assembly of the Westinghouse internals, 7 holes were drilled (depth : 19 mm). Two core drilling cutters were needed and the whole operation (including the positioning of the machine, the rotation of the piece to present it correctly in front of the cutter, the cutter exchange, ...) took 6 hours, required 27.4 man*hours and gave 0.119 man*mSv.

4.3 Unbolter

Unbolting is a very useful technique because dismounting allows to avoid a cut or allows to separate an assembly into subassemblies which have a simpler geometry and are therefore much easier to dismantle afterwards.
When the bolts are placed vertically and the accessibility to their top is possible from a footbridge, the only remaining problem concerns the bolt safety system used. The type of safety system mostly used at BR3, consisting of bended shroud around the bolt head, allowed to place standard sockets on the bolt top. The impact unbolter allowed to break the safety, by the shocks produced by the machine.
For the dismounting of the Rod Shroud Support Plate of the Vulcain internals, standard nuts (M18) were secured by a transversal safety pin (diameter :4 mm). The unbolting was carried out using reduced distance between the operators and the pieces (2 m water), heavy connecting rod and socket to transmit the power and a heavy pneumatic unbolter. Each safety pin broke after a sequence of bolt-unbolt motions.
When the bolts are placed top down and if the bolts heads are accessible, a hydraulic unbolter handled at the extremity of a long handling tool can be used. The positioning operations have to be made with remote underwater camera viewing.

4.4 Reciprocating saw

This reciprocating saw is based on a mechanical principle. The tool is pneumatically driven and has been adapted to work underwater. The "forward and back" motion of the saw blade represents the cut movement. The feed has to be given by specific supporting device.
That tool presents a lot of important advantages :

- the saw blade is fastened only at one extremity, so the accessibility at both sides of the piece to cut is not needed ;
- the saw blade length can be up to 600 mm, so it is possible to go very deep into the piece to perform a cut ;
- the saw blade has a thickness of about 2 mm, so the kerf is thin and the sawing does not produce too much secondary waste (only small chips).

This technique was used to cut all the tubes penetrating the instrumentation collar of the Vulcain internals, to cut the collector piece of the Vulcain Upper Core Support Assembly and to achieve horizontal cuts on the Westinghouse core baffle, where the strip to be cut was too deep to be reached by the circular saw.
This tool was also mounted on a frame to cut high activated tubes such as the moderator tubes, rod shroud tubes, control rods ...

4.5 Electro Discharge Machining

E.D.M. (EDM) stands for Electro Discharge Machining.
This technique is used in workshop to print the shape of an electrode into metallic material; it is also possible to cut metal by using thin plate electrodes.
During the comparison of cutting techniques on the thermal shield, we showed that EDM was not the best solution to cut thick material. It produces too much secondary waste and the process is too slow.
Nevertheless, for some specific applications, EDM can help dismantling : the biggest advantage of EDM is that almost no force is applied between the electrode and the workpiece and that almost any shape can be given to the electrode. In the BR3 project, the EDM was also used to dismount a plate from equipments which were bolted to it underneath. The bolts were not accessible by standard dismounting tool and we decided to cut the bolts by perforating the plate. Unfortunately, some bolts were placed just under reinforcement ribs of the plate. An oblique perforation EDM system was developed, tested and succesfully applied.
The EDM part of the task concerned 38 bolts; twenty were EDM-perforated vertically and eighteen oblique. Cutting a bolt vertically took in average 0.39 working day (one shift/day) as total operational time and resulted in an average dose uptake of 0.058 man*mSv. Cutting a bolt in an oblique way took in average 0.8 working day and resulted in an average dose uptake of 0.121 man*mSv.
The use of EDM is not recommended as a cutting technique for dismantling thick reactor internals, but, for some *"surgery operations"*, its flexibility can be a definite advantage. Any surgery EDM work needs a lot of developments and tests.
For the dismounting mentioned hereabove, we developed specific electrodes and the positioning system of the EDM-head had to be very precise.

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5. The dismantling after 30 years cooling down period

One of the objectives of phase 2_{bis} of the project was the comparison of the direct dismantling of internals with the deferred dismantling, after 30 years of cooling down period.
Before starting the dismantling of the cooled down internals (so-called "Westinghouse" internals), measurements of the radiation dose rate were made as well as a theoretical calculation of the dose rate and activity level when unloading.
This theoretical approach was intended to determine the specific activity when unloading to be able to compare the two operations.
The theoretical approach, based on neutronic calculations and activation computations gave quite surprising and interesting results. Indeed, they showed that the build up of the ^{60}Co saturates quite rapidly, after only three or four working campaigns and tends even to decrease after the first four working campaigns, depending on the initial cobalt concentration in the metal.
This is mostly due to the decrease in ^{59}Co content due to neutron absorption, to the natural decay of the produced ^{60}Co but also to the disappearing of the generated ^{60}Co by radiative capture under neutron irradiation. The combination of these factors give an evolution of the ^{60}Co activity and explains the quite high level of radiation still present on the "Westinghouse" internals.
When calculating the activity level of the internals at the time of unloading, based on the activity measurement done on the swarfs of the internals and on the decay law of the ^{60}Co, one gets the table here below (table XI).
In this table, the specific activity of the Vulcain internals, being irradiated for 21 years, is given for comparison.
One can see that the ^{60}Co specific activity when unloading was indeed of the same order of magnitude. Care must be paid to the initial concentration of cobalt in the steel : this data is not well known for BR3 and can have an important influence.

Level vs midplane	Westinghouse internals 2 operating campaigns		Vulcain internals 9 operating campaigns	
	in 1995 after 31 years cooling down	in 1964 when unloading	in 1994 after 7 years cooling down	in 1987 when unloading
- 413 (baffle)	3.454	203.43		
- 213 (baffle)			125.0	313.9
+ 311 (barrel)			19.0	47.7
+ 497 (barrel)	0.565	33.28		
+ 801			1.2	3.0
+ 929	0.073	4.30		

Table XI : Comparison of the specific activity of the two internals

Nevertheless, the important feature for the comparison of the 2 dismantling operations was that the Westinghouse internals showed representative and similar activity contents when unloading to be able to compare the 30 years deferred dismantling and the almost direct dismantling on the same base.
After a cooling down period of 30 years, the BR3 Westinghouse biggest internal, the *"Lower Core Support Assembly" (LCSA)* presented in water at its outside surface at the midplane a dose rate of about 1.7 Sv/h. Inside the LCSA the dose rate rose up to 7 Sv/h. With such a high dose rate level, hands-on dismantling or dry cutting without important shielding is impossible. Therefore, underwater dismantling was the selected method.
For the dismantling of the Westinghouse LCSA, the configuration of the pool where the cuts were carried out was precisely the same as the configuration used for the immediate dismantling of the Vulcain internals (about 7 m of water). The reactor core mid plane was during the whole cutting campaign under a minimum of 4 m water (even more). Considering that the "half depth" of water is about 12 cm for ^{60}Co, the gamma radiation reduction factor is about **10^9**. The activity reduction factor due to the 30 years cooling down, which is about 50, is negligible compared to this last factor. No significant influence of the cooling down period could then be

expected for the direct radiation uptake.
Another important point was the detection of trapped swarfs :during tools and equipment maintenance (out of the water), the detection of high dose rate swarfs is easier than low dose rate swarfs. The high dose rate swarfs, easily detected and located, are then directly eliminated by cleaning while the low dose rate swarfs can sometimes only increase a bit the ambient dose rate. This paradoxical effect can lead to higher dose uptake when working on lower active pieces than on high active pieces.
When comparing the dose uptake for the two sets of internals (see table XII hereafter) the values are higher for the "Vulcain" internals than for the "Westinghouse" internals (87 µSv/dm² and 40 µSv/dm²). This difference is not due to the internals specific activity (shielding by water) but to the change in the ambient dose rate on the reactor operator deck.

Technique used	Cut length	Cut section	Time for cutting & preparation	Effective cutting speed (*)	Dose uptake	Dose/dm²
	[m]	[dm²]	[hours]	[dm²/hr]	[man*mSv]	[mSv/dm²]
Phase 1 : Thermal Shield						
EDM	7.1	54	424	0.13	14.6	0.271
Circular saw	12.5	95	192	0.49	8.3	0.087
Plasma	15.0	114	136	0.84	9.2	0.081
Total	34.5	263	752	0.35	32.1	0.122
Phase 2 : Vulcain Internals						
Circular saw	43.4	95	384	0.25	9.1	0.096
Band saw	60.7	217	872	0.25	17.9	0.083
Total	104.1	311	1256	0.25	27.0	0.087
Phase 2_{bis} : Westinghouse Internals						
Circular saw	30.2	63	250	0.25	3.8	0.060
Band saw	58.8	161	359	0.45	5.3	0.033
Total	89.0	224	609	0.37	9.1	0.040
Whole operation						
Total	227.6	797.9	2617.2	0.30	68.1	0.085

(*) Effective cutting speed : it is the time between two cuts (installation, cut, cut piece removal and clamping adaptation).

Table XII : Summary of the results of the reactor internals dismantling

Indeed, between phase 2 and phase 2_{bis}, different radiation sources present in the operation area were removed. This removal led to decrease the ambient dose rate from 10-15 µSv/h during phase 2 to 4.5 µSv/h during phase 2_{bis}. This difference explains completely the decrease in the dose uptake experienced for phase 2_{bis}, which is not related to the workpieces activity.
Regarding the waste production, no significant advantage of the cooling down period could be observed. If we compare the immediate dismantling of the Vulcain LCSA and the dismantling of the Westinghouse LCSA (after 30 years), the waste category transition levels are almost the same. The only difference concerned one ring : on the Vulcain internals, the transition from *"LLW to MLW"* occured around the upper part of the hot leg; on the Westinghouse LCSA, it occured under the hot leg. So the gain we made on the Westinghouse internals was the category change of 283 kg of material from MLW to LLW. Concerning the production of swarfs, no gain occurred because no specific (complicated) selection was made during the swarfs recuperation.

The main conclusion is thus that no significant advantage can be drawn from waiting for 30 years before dismantling highly radioactive pieces. This seems to be valid for the radioprotection point of view as well as for the waste production.
Regarding the similar activity of both sets of internals at the time of definitive unloading, this conclusion is based on sound comparison results.

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The BR3 dismantling operations and related techniques

Dismantling of highly radioactive reactor internals

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6. Summary and comparison of the cutting processes

The results produced during the whole cutting campaign of the BR3 internals are summarized in the next table.

Technique used	Cut length	Cut section	Time for cutting & preparation	Effective cutting speed (*)	Dose uptake	Dose/dm²
	[m]	[dm²]	[hours]	[dm²/hr]	[man*mSv]	[mSv/dm²]
			Phase 1 : Thermal Shield			
EDM	7.1	54	424	0.13	14.6	0.271
Circular saw	12.5	95	192	0.49	8.3	0.087
Plasma	15.0	114	136	0.84	9.2	0.081
Total	34.5	263	752	0.35	32.1	0.122
			Phase 2 : Vulcain Internals			
Circular saw	43.4	95	384	0.25	9.1	0.096
Band saw	60.7	217	872	0.25	17.9	0.083
Total	104.1	311	1256	0.25	27.0	0.087
			Phase 2_{bis} : Westinghouse Internals			
Circular saw	30.2	63	250	0.25	3.8	0.060
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Table XII : Summary of the results of the reactor internals dismantling

It is important to remember that the work piece of phase 1 was the thermal shield (a quite simple cylinder of 1.5 m diameter and 3 inches thickness, or 76.2 mm). For phases 2 and 2_{bis} the workpieces had a more complex geometry with different plate thicknesses, different diameters, protrusions, etc.
The next table (table XIII) gives an overview of the secondary waste volume produced during the job. The waste volume is the sum of :

- the volume of the coarse filters from the local filtration unit installed in the reactor pool to collect the swarfs directly at the source ;
- the volume of the fine filters from the local filtation unit installed in the reactor pool ;
- the volume of the fine filters from the filtration unit of the plant ;
- and, finally, the worn tools of the different equipments (saw blades, electrodes, ...).

Technique used	Cut section	Amount of metal released	Secondary waste volume	Ratio waste vol/ cut section
	[dm²]	[dm³]	[dm³]	[dm³/dm²]
		Phase 1		
EDM	53.87	3.77	377.2	7.0

Circular saw	94.87	3.79	130.3	1.4
Plasma	114.15	12.56	781.6	6.8
Total	262.89	20	1289.1	4.9
		Phase 2		
Circular saw	94.55	5.67	199.5	2.1
Band saw	216.65	4.33	292.7	1.4
Total	311.20	10.01	492.2	1.6
		Phase 2_{bis}		
Circular saw	63.24	3.79	111.2	1.8
Band saw	160.60	3.21	105.5	0.7
Total	223.84	7.01	216.7	1.0
		Pool cleaning		
Pool cleaning	-	-	153	-
		Whole operation		
Total	797.93	37.02	2151.0	2.7

Table XIII : Overview of the secondary waste volume

If we compare the results obtained with the mechanical technique, it is important to note that the thickness of the circular saw blade or of the band saw blade varied during the different phases : 4 mm (phase 1), 6 mm for the circular saw blade and 2 mm for the band saw blade (phases 2 and 2_{bis}).

Moreover, during phase 2, the fine filters of the local filtration unit were not used during a part of the cutting campaign. The total volume of these filters is about 116 dm³. Thus, the results obtained during phase 2 are of the same order of magnitude than those of phase 1 and the results of phase 2_{bis} are better than the others (better filling of the coarse filters).

At the end of the whole project a deep clean up of the reactor pool bottom has been carried out. It resulted in the production of 9 supplementary strainers (about 153 dm³). These waste can not be related to any one of the project phases but has to be added to the overall secondary waste volume production. Another lesson learned during the two last phases is the improvement of the prediction of the necessary worktime : table XIV illustrates this.

		Phase 2			Phase 2_{bis}		
Technique	Parameter	Predicted	Observed	Ratio observed / predicted	Predicted	Observed	Ratio observed / predicted
Circular saw	Cut section [dm²]		94.55			63.24	
	Time for cutting [hrs]	312	384	1.23	196	250	1.28
	Manpower [man-hours]	930	1 444	1.55	804	930	1.16
	Dose uptake [man-mSv]	13.85	9.05	0.65	6.01	3.77	0.63
	Dose uptake/dm² [mSv/dm²]		0.096			0.060	
Band saw	Cut section [dm²]		216.65			160.60	
	Time for cutting [hrs]	504	872	1.73	426	359	0.84
	Manpower [man-hours]	1 443	2 524	1.75	1 725	1 347	0.78
	Dose uptake [man-mSv]	21.65	17.94	0.83	8.68	5.28	0.61
	Dose uptake/dm² [mSv/dm²]		0.083			0.033	
	Cut section [dm²]		311.20			223.84	
	Time for cutting [hrs]	816	1 256	1.54	622	609	0.98
	Manpower [man-	2 373	3 968	1.67	2 529	2 277	0.90

	hours]						
TOTAL	Dose uptake [man-mSv]	35.50	26.99	0.76	14.69	9.05	0.62
	Dose uptake/dm² [mSv/dm²]		0.087			0.040	

Table XIV : Comparison of observed and predicted manpower and dose uptake

During *phase 2*, the total observed time in hours and man-hours is about 50 % higher than predicted. This difference comes from the operations that were not taken into account due to the lack of experience for all problems occurring during the work.
Afterwards this was taken into account and estimated to be about 30 % of the theoretical cutting time. For the manpower, the difference is due to the same reason but also to the continuous presence of partners delegates during the whole phase. You can note the divergence between the results of the circular saw and the band saw. In fact, during phase 2, a problem occurred with the band saw and it was stopped during 13 whole days on a total of about 109 working days (12 % of the working time).
During *phase* 2_{bis}, the total predicted and observed hours and manpower are in good accordance. For this prevision several factors were taken into account :

- a supplement of 30 % for all problems occurring during the work ;
- the experience gained during phase 2 ;
- some pieces were dismounted and not cut.

During this phase, a difference in working time and man-hours appeared between the two machines. Here, due to a problem with the circular saw, the machine was stopped during 79 hours on a total working time of 250 hours (32 % of the working time).
Although the optimisation of the cutting parameters is important to have a reliable technique, it is important to try to reduce the times allocated to the other operations (installation, saw blades exchange, cut pieces evacuation, filtration devices and filter exchange, etc.).

Comparison of two dismantling methods

The comparison between the two sets of internals shows that there are two similar cuts situated at the level of the bolted link between the core support plate and the core barrel; the dismounting methods are different for both internals.
The circular saw has been chosen to cut the "Vulcain" internals, while an unbolting device to dismount the "Westinghouse" internals seemed the most appropriate. This difference in the dismantling approach comes from the fact that for the "Vulcain" internals, some pipes were present through the core support very far from the core barrel. Therefore, it was necessary to use the circular saw to cut it.
To balance the different results obtained, it is interesting to compare this one with a normal cut result (without any bolted links) : the next table gives the comparison.

	Type of operation	Total cutting time	Manpower	Dose uptake	Dose rate equivalent
		[hours]	[man-hours]	[man-mSv]	[mSv/h] *
Phase 2	Circular saw	108	408	1.63	0.004
Phase 2_{bis}	Manual unbolting	26	94	0.37	0.004
Reference : Phase 2_{bis} / Standard cut	Circular saw	36	135	0.73	0.005

Table XV : Comparison of two dismantling methods for bolted assemblies

* Note : The dose rate equivalent is a calculated mean dose rate, based on the actual man-hours and mSv uptake.
Where it is possible, unbolting is the best solution : the total operation time is shorter; direct access and direct vision facilitate the work. Compared to a standard cut without bolted links, the two operations gave similar results. In addition, the use of a standard machine reduced the operational total cost.

Summary of the main results

The different parts of the project were described in the preceding chapters, including their detailed results. As a summary, some concluding results can be pointed out.
First, for the *pre-dismantling decontamination* of the primary loop, this operation has shown its advantages :

- it reduces significantly the overall dose rate around the primary loop and in the work area; the total dose uptake balance shows a significant advantage to perform this operation ;

- even for the dismantling of the internals, it allows to change the waste category by removing the hot spots due to the contamination ;
- the total amount of radwaste produced by this operation can be limited, but it is difficult to forecast and to control precisely the amount of dissolved ions produced by the operation (non uniformity of the crud deposits).

Moreover, some further lessons can also be drawn : indeed, it is important to execute the operation quite soon after the plant shutdown to be able to reuse the plant installations and equipment for the operation.
For the *dismantling of the reactor internals*, different cutting techniques were compared. This has shown that, for highly radioactive internals, the mechanical cutting methods can be used easily and gave very good results concerning the produced waste, the contamination of the environment, the cutting duration and the dose uptake.
Different mechanical cutting methods were used and compared, and each of them showed their advantages and drawbacks. For the dismantling of large pieces presenting complicated shapes, the band sawing has proved to be very effective.
Moreover, carrying out the operations under water prevented to produce a high dose uptake to the operators.
The presence of two sets of internals, having undergone different cooling down periods (7 years and 31 years), has shown that the difference in activity due to the natural decay of ^{60}Co does not influence significantly the operator's dose uptake. Also, the cooling down period of 31 years of the first set was not sufficient to imply any improvement or any important advantage for dismantling these pieces.
Another interesting result was that after only two years of operation, the first set of internals presented at the time of unloading, an activity of the same order of magnitude as the one having been irradiated for 21 years.

CONCLUSIONS

This pilot decommissioning project, supported by the European Commission in the framework of its five year RTD programme on the *"decommissioning of nuclear installations"*, has brought interesting results for future decommissioning of nuclear power plants and installations.
Moreover, as a *"pilot project"*, the BR3 project allowed the different European contractors and partners to build up experience in this field, which is now developing quite fast.
The project has associated industries from different countries in the European Union and got international reputation through its multinational character. Moreover it was the first one dealing with the dismantling of a Pressurized Water Reactor (PWR), which is one of the reactor types the most widely spread in Europe and in the world.
Research and Technical Developments have been carried out throughout the project and allowed to get interesting results developed in the report. Nevertheless some Development works and even Research are still needed to continue improvement of the operations, the procedures, the technologies, etc ..., in order to reduce again the dose uptake and the waste production. This is probably important for an activity rapidly growing in the nuclear sector, and even for the nuclear industry itself.

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Wastes Management at BR3

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1. Introduction

Dismantling a nuclear facility generates an enormous amount of radioactive material, which must be evacuated one way or another.
The base philosophy in this evacuation process is to *minimize the volume of radioactive waste*.
Furthermore, during the evacuation process, the material has to satisfy an important condition, namely that *its traceability is assured*. Therefore the *material flow* is a well defined procedure which guides the material from the dismantling until definitive evacuation.

Thus the material which is dismantled undergoes some steps :

- cutting, sorting and identification ;
- temporary storage ;
- treatment ;
- characterization ;
- temporary storage ;
- evacuation.

2. Cutting, sorting and identification

After the dismantling, the large pieces are transported to a cutting-chamber. This is a closed ventilated booth where the dismantled material is cut in smaller pieces.
For the cutting up of the pieces, different techniques are used. The evacuation way can influence the choice of the cutting technique.
Following cutting techniques are used :

To be decontaminated	Waste / Melting
grinding	plasma cutting
nibbling	oxy-acetylene cutting
band saw	
reciprocating saw	

Plasma and oxy-acetylene cutting are not used for material which has to be decontaminated, because while cutting, the contamination will (could) be enclosed by the slag on the pieces.
The band saw and the reciprocating saw are used for the cutting of tubes / pipes or "massive" pieces.
The nibbling technique is used for the cutting of tanks : it is a relatively fast technique but has the disadvantage that it leaves sharp ends on the cut pieces.
After the material is cut into smaller pieces, it is sorted following the different evacuation ways.

Waste		Melting		Treatment manual decontamination wet-sandblasting chemical decontamination scabbling	
compactable	non compactable	re-use in nuclear environment	free release	unconditional free release	free release after melting

Each evacuation way has its own acceptance criteria :

- **Waste**

	compactable	non-compactable
waste package	200 liters drum	400 liters drum
max. length of the piece	0.80 m	0.95 m
max. weight of the piece	50 kg	no limit
max. thickness of the piece	20 mm	no limit
max. weight of aluminum	10 kg	not allowed
max. weight of PVC	20 kg	not allowed
lead	not allowed	not allowed

- **Melting**
  Metal which is technically or economically not possible to decontaminate will be sent to a melting facility. For example tubes and valves with a diameter < 1", thin plates and beams, small pieces like bolts, small infrastructural elements, ...

- **Treatment**
  The selection of the material to be treated depends on two conditions :
  - it has to be technically possible and economically defensible ;
  - it has to be possible to demonstrate that material fulfills the free-release criteria.

The following table gives a global idea of the sorting criteria.

	geometry	contamination	examples
manual decontamination (cleaning, polishing)	simple	low level contamination	electrical cables, infrastructural elements
wet-sandblasting	simple good accessibility small surface / weight ratio	fixed contamination up to 5000 Bq/cm²	large infrastructural elements, heavy solid pieces
chemical decontamination	simple to complex	fixed contamination up to 35000 Bq/cm²	tubes and valves with a Ø > 1", pumps, tanks
scabbling			concrete blocs

During the sorting, the batches are created.
A batch can be considered as a group of material which will follow the same evacuation way. The material can be grouped in :

- 200 liters / 400 liters barrels if it concerns waste or material to be melted.

Occasionally it can also be material to be free released if the activity is homogeneously distributed (which is for example the occasion for dust, coming from the decontamination of concrete blocs.

- 300 liters plastic containers if it concerns material to be treated
- or it can be a single piece, if it concerns large pieces for melting.

Every batch is identified by a label sticked to it, on which is written down a unique identification number, the contents and the weight of each category of material which is in the batch.

3. Temporary storage before characterization or treatment

When the batches are completed, they are stored in expectation of treatment campaigns or characterization.

▲ 4. Treatment

To minimize the amount of radioactive waste, the material can be treated in different ways.
A lot of the dismantled material can be treated manually, because its surfaces are only slightly (or even not) contaminated, and the contamination is not fixed at the surface. This material, mostly infrastructural elements, electrical cables, instrumentation, is cleaned with water and detergent.
Pieces with a higher degree of contamination, or where the contamination is fixed (in the rust or in the paint), can be treated by wet-sandblasting (water with sand at high pressure, called ZOE) or "aggressive" chemical decontamination (MEDOC).

5. Characterization of the material

5.1. Material for free release

After the decontamination of the material, it is measured with hand held monitors (with a measuring surface of 25 cm² and 200 cm²).
The condition here is that the whole surface of the material must be accessible for the monitors; if this is not the case the material will be rejected.
After this measurement, the material that fulfills the free release conditions (surface contamination < 0,4 Bq/cm² for beta-gamma contamination and < 0,04 Bq/cm² for alpha

contamination), will be stored in expectation of a second measurement. This measurement will be done by another person and with another type of monitor.
The storage time between two measurements can be 3 months if it concerns material which underwent a chemical decontamination. This is to be able to detect radioactivity released by the "*sweating*" of the material.
The results of the free-release measurements together with the history of the material is collected in a "*free-release dossier*" which has to be approved by the head of the "*Health Physics Department*" before definitive free-release.
As mentioned earlier, material which doesn't allow a 100 % surface accessibility will not be accepted to be measured. For a lot of material at BR3, this is the case, mainly tubes and valves. For this material, there is still no accepted procedure for the measurements.

5.2. Material for waste and melting

This category of material is mostly put into barrels (200 liters / 400 liters) and is characterized by a gammaspectrometrical system. The large pieces (for melting) are characterized by sampling.

6. Temporary storage after treatment and characterization

After the material is characterized, it is stored while it awaits definitive approval for free release, or evacuation.

7. Administrative follow up

During the evacuation process (which begins right after the dismantling and ends with the evacuation), a close follow up of the material must be done to assure the traceability of the material.
This means that at all times, the batches must be well identified, the contents must be well known as well as the status of it (awaiting decontamination, already measured for free release, ...). This methodology must allow to evacuate the dismantled material in an as much as possible controlled manner.

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▲

Waste management

Study of Dismantling Strategy

This activity is performed under the Fusion Long Term Programme on Waste Management, task TSW2D4. For further information, please contact Vincent Massaut (vmassaut@sckcen.be).

OBJECTIVES

Present-day design of nuclear installations includes planning for decommissioning. This covers not only technical capabilities, but also a financial analysis. In order to cover the nuclear liabilities proceeding from the use of nuclear facilities, a cost estimate for the decontamination/dismantling phase must be calculated, allowing for suitable provisions to be made during the lifetime of the facility. This applies also to fusion reactors, as they will similarly generate radioactive material. It is therefore necessary to look at the impact of different decommissioning strategies on the management of activated fusion materials. For this cost estimate, a detailed analysis of decontamination/dismantling strategies and technical alternatives has to be carried out.

ACHIEVEMENTS

The past ten years SCK•CEN has gained a lot of know-how and expertise during the decommissioning of a small fission reactor. The BR3 reactor is a 10 MW_e pressurised water reactor (Westinghouse), used for more than 25 years, as fuel test and training facility for NPP-operators. It was shutdown in 1987 and selected as one of the four pilot dismantling projects by the EU for its R&D programme on Decommissioning of Nuclear Installations. During several years, different decommissioning activities have been carried out. These activities which are similar or could be of interest for the dismantling of a fusion reactor have been analysed and studied. A first report was issued at the beginning of 2000.

Comparison between immediate and deferred dismantling

On BR3, we dismantled two sets of high active internals which had different decay times (5 and 30 years). No significant radiological, technical or economical profit was gained by deferred dismantling because we had to use the same techniques (remote mechanical cutting underwater) due to the still remaining high dose rate.

Comparison between different techniques for cutting highly activated components

In a PWR environment, it is common to use underwater remote techniques during operation. Under dismantling conditions, it is also advantageous to use water as a shielding during the cutting of highly activated parts. We compared thermal and mechanical techniques from technical, radiological and economical points of view. We strongly support remote operated mechanical techniques (milling cutter, circular saw, band saw, hydraulic shears...) mainly because the lower volume and easier to collect secondary waste produced, the better underwater visibility easily maintained by simple filtration and purification systems, and the acceptable cutting speed achieved.

For the dismantling of contaminated components, we have reduced the dose rate by a full system decontamination of the primary system. Hands-on techniques can then be used at most of the locations. Here also, to avoid contamination spread, mechanical cutting techniques, with low aerosols production, are to be preferred.

In a fusion reactor, remote handling is already available during the operation of the machine. This equipment can also be used during the dismantling phase of the highly activated parts inside the cryostat. The fusion reactor environment, however, is quite different from that of a PWR. A good compromise could be to combine easy to deploy techniques such as plasma cutting, laser cutting or high pressure water (with or without abrasives) for the on-site cutting of large pieces, followed by size reduction in a separate hot workshop, using either the same techniques or other mechanical approaches. For the hot workshop, a comparison should be made between underwater operation or a classical hot cell.

Lessons learned and application to the fusion reactor

- Thorough environmental dose characterisation is important not only to detect hot spots but also to estimate low activation levels.
- When dismantling is started, all infrastructure for the materials management should be already in place.
- Decontamination techniques are to be used to reduce the dose rate levels, allowing recycling, either in the nuclear world, or with unconditional clearance. Recycling of high active materials is however not applied up to now in the nuclear industry. Melting of active metal waste was for instance limited up to now to relatively low active pieces.
- Recycling techniques for low activated concrete are available.

FUTURE WORK

At the end of 2000, the following aspects will be reported:

- Lessons learnt from the actual decommissioning on the technical, economical and radiological point of views and application to the fusion reactors.
- Analysis of the complex management of the decommissioned materials and of the various existing evacuation routes.
- First analysis of the recycling issue, from the technical and licensing point of views. We will examine the recycling of valuable high active materials inside the "fission field" as well as the recycling in the nuclear world or, more generally, in the industrial world.
- The presently developed methodology for the calculation of the fission dismantling costs will be presented with the possible aim of applying it to fusion reactor dismantling.

REFERENCES

[1] V.Massaut, L.Denissen, M.Klein, European Fusion Long Term Programme - Safety and Environment, Task S5.2 Management of activated material - Decommissioning Strategies, Intermediate Report, SCK.CEN R-3445, April 10, 2000.

[2] M. Klein, J. Dadoumont, Y. Demeulemeester, V. Massaut, Experience in Decommissioning activities at the BR3 site, IAEA Technical Committee Meeting on Fusion Safety, Cannes, 13–16 June 200 (proceedings in preparation).

Evacuation of the BR3 spent fuel

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1. General information on the fuel

1.1. Introduction

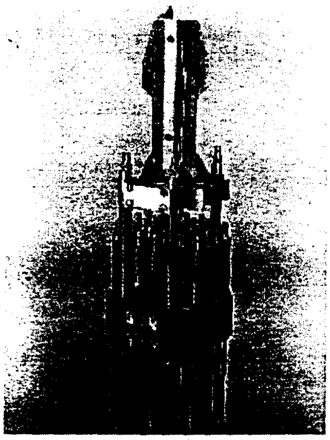
During its exploitation period, the BR3 reactor has had eleven successive core loadings. Each core loading consisting out of 73 fuel elements except the first and the second (32 assemblies), most of the fuel elements were irradiated during several cycles.
The spent fuel of the first and second core was already returned to the owners in an earlier period, this was also the case for several experimental assemblies coming from different countries. Nevertheless, the remaining spent fuel has to be removed out of the pond in order to proceed with the decommissioning of the BR3 power plant.

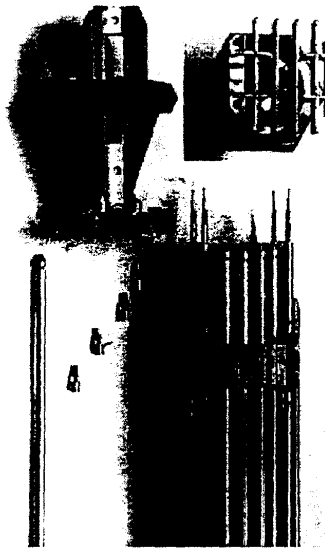
As mentioned before the BR3 power plant was used as a test bench for prototype fuels. During its life time, this led to the presence of a large variety of spent fuel, which is stored in the spent fuel pond. The assemblies used in the 9 latest core loadings have a hexagonal cross-section. There are 5 different hexagon fuel assembly types irradiated.
Table 1 gives an overview of the different dimensions and characteristics of the fuel.

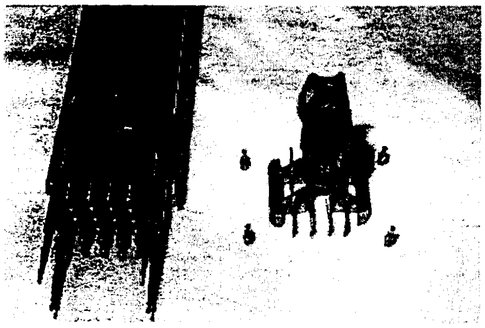
Type	Rod Ø(mm)	Cladding material	Overal length (mm)	Isotopic composition (enrichment)				Ratio-rods U/Mox
				U-rods ^{235}U (%)	Mox-rods		Gd_2O_3 (%)	
					^{235}U (%)	Pu_{fiss} (%)		
Z	8.7	Zircaloy4	1235	7.1	3	3.1 - 5.0	-	36/0 → 10/26
Gf	9.4	Zircaloy4	1136	6.4 - 8.6	-	-	-	28/0
IF	8.5	Stainless steel	1251	-	0.7	6.4	-	0/37
G	10.7	Zircaloy4	1136	5.1 - 8.6	3	3.7 - 7.0	1.35	20/0 → 0/20
Go	9.5	Zircaloy4	1136	5.0 - 8.3	0.3 - 0.7	3.7 - 10.3	3.0 - 7.0	28/0 → 0/28

Table 1. Overview of the different fuel types

The spent fuel is stored in storage racks, which can contain 203 fuel assemblies of which 90 % is presently used. The storage racks are foreseen of Cadmium cladded storage positions.







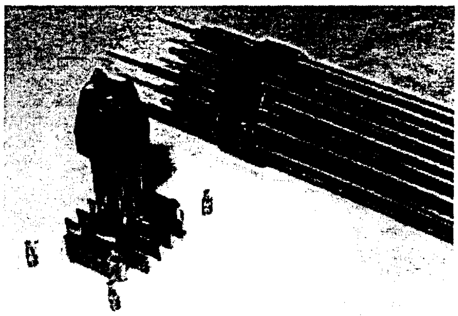


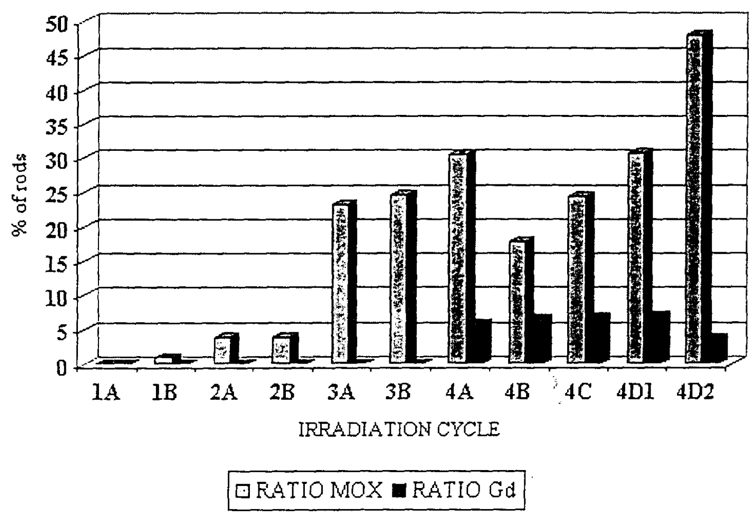
Figure 1. Bundle with head on

Figures 2 to 4. Bundle with its head (side and top) screws

1.2. General data on the fuel to be evacuated

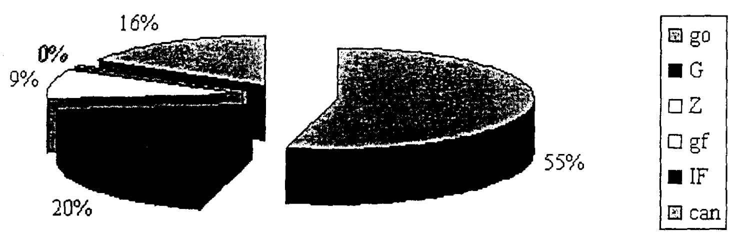
- Ratio of Mox- and Gd-rods in a reactor core loading
  The diagram shows the ratio of Mox- and Gd-rods per irradiation cycle.

PROPORTION OF MOX AND Gd-RODS per CYCLE



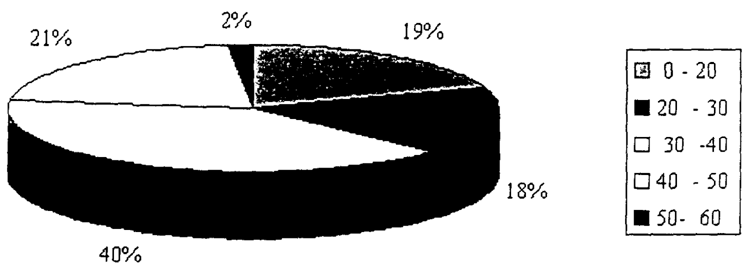
- Number of fuel assemblies per type
  The pie-diagram shows the different types of BR3 fuel to be evacuated. The can-type (canister) is used evacuation of non-intact fuel pins. There is only one fuel assembly of the type gf and type IF to evacuate

DIFFERENT TYPE OF FUEL ASSEMBLIES



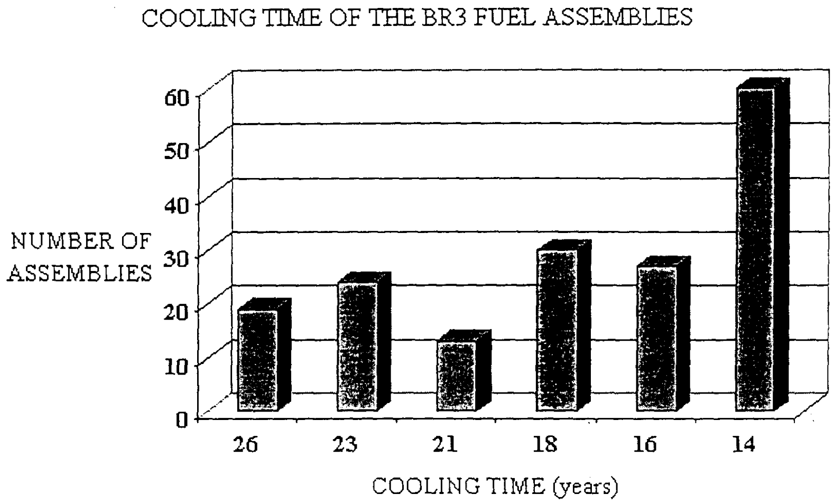
- Burn up distribution

BURN UP DISTRIBUTION

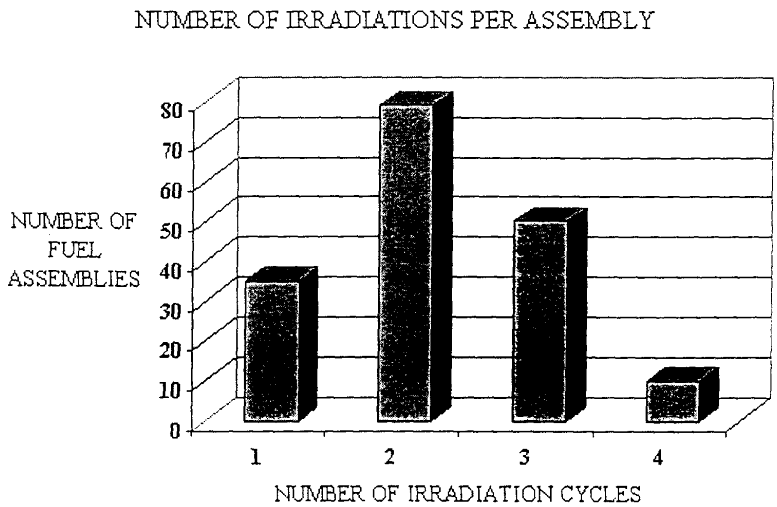


- Cooling time of the fuel assemblies
  Most fuel assemblies have been stored in the spent fuel pond over a period of 14 years. But the diagrarr

that the maximum storage time of some irradiated fuel assemblies can raise up to 26 years.



- Number of irradiation cycles per assembly
  Most fuel assemblies have been irradiated during two cycles, some have been irradiated during four cycles.



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2. Existing Storage facility

2.1. Storage of the spent fuel

The storage of the BR3 spent fuel is at this moment performed in the deactivation pool of the nuclear power plant. The so called storage well is a pond constructed out of heavy concrete and cladded with stainless steel. The dimensions are 6.1 x 6.17 x 11.87 m) and can be filled with water to a level of 10.45 m which represents a water-shielding against irradiation of more than 7 meters.

Figure 5. Top view of the storage pond BR3

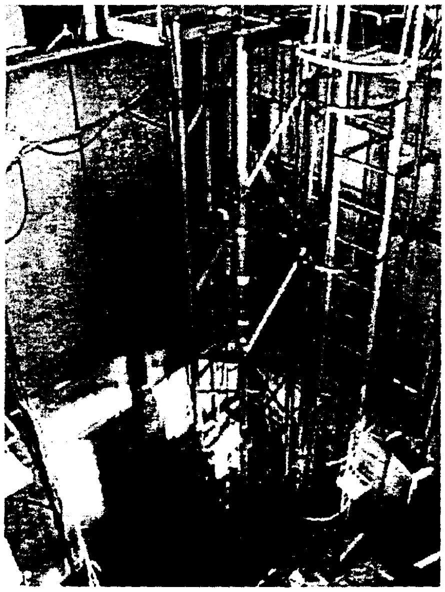
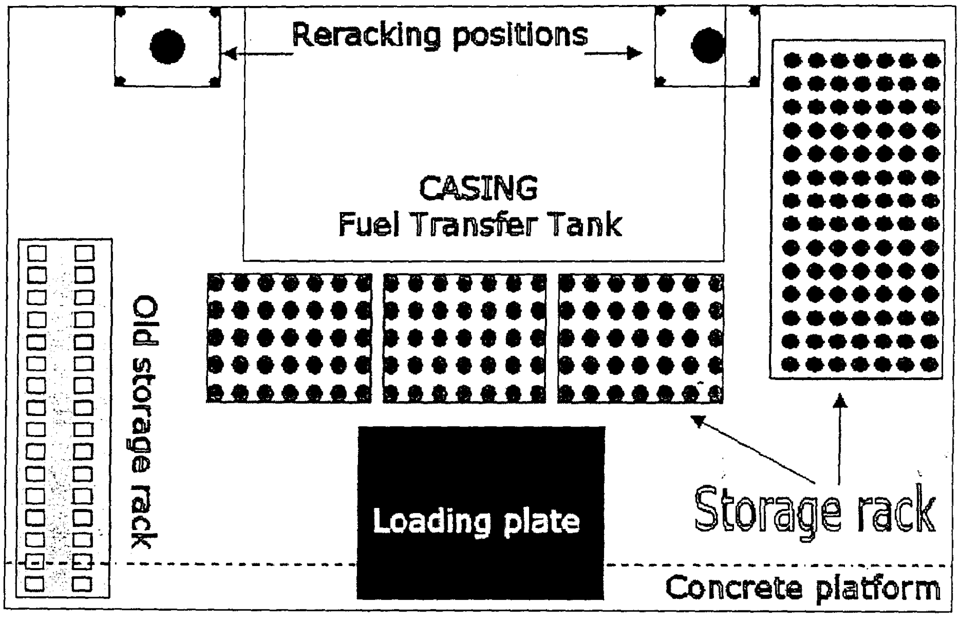
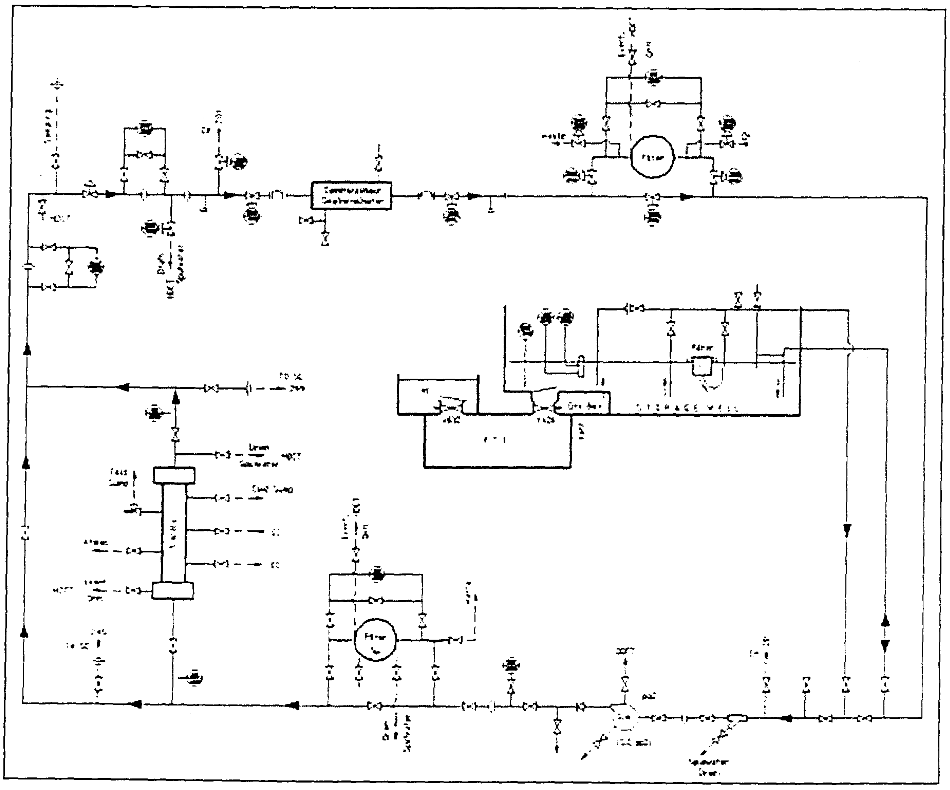


Figure 6. View of the dismounting installation

The fuel assemblies are stored in storage racks. These racks consist of storage tubes which can store one assembly; the tubes are seperated by neutron absorbing material (Cd-screens).



The storage well water is continuously filtered and purified (decontaminated) by a filter- and resin-system. In the past the circuit had also the possibility to cool the water, but due to the long cooling time of the fuel assemblies the residual head production is neglectable. The contamination level, the pH value and the conductance of the water are checked on a regular basis.



2.2. Handling of the spent fuel

The following manipulations are possible on fuel assemblies with the installations and equipment foreseen in the storage well :

- Transfer of fuel assemblies in the storage well (to other storage position, transport container or reactor building);
- Dismounting type go assemblies, to exchange rods between type go assemblies;
- Leak tightness test of fuel assemblies by Wet Sipping;
- Loading of a transportcontainer;

Therefore, the following main equipment is foreseen :

- A crane which can handle a charge of 30 tons, it is recently equipped with a double brake system to meet the latest safety standards. The bridge is always used in combination with long handling tools to perform manipulations on the fuel assemblies;
- Two positions for consolidation or dismantling fuel assemblies of the go type;
- For internal transports (on the SCK·CEN site) the TN-6 container is used :

- Two Wet Sipping installations :

2.3. Surveillance

There is a continuous surveillance of the spent fuel by SCK·CEN staff and by a recording system of IAEA and Euratom. Each 3 months a physical check of the inventory of the spent fuel pond is performed by inspectors of IAEA and Euratom.

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3. Description of the project

3.1. Overview

The fuel of the first four irradiation cycles was transported to the owners during an earlier period. At this moment there are still 175 assemblies stored in the BR3 storage pond. The non-intact fuel pins are assembled in welded canisters at the BR2 pool.

3.2. Different options

For the evacuation of the BR3 spent fuel several options were studied :

- Modernisation of the BR3-pond to enlarge the storage period
- Reprocessing of the spent fuel
- Dry storage of the spent fuel
  - At the reactor site
  - Away from the reactor site

3.3. Options dropped after a first study

Two of the three options were deleted after a first study with the following conclusions :

- Modernisation of the BR3-pond can not be justified for a plant in decommissioning
- Reprocessing because :
  - This solution was more expensive (3x) than other options
  - The Mox fuel, irradiated in the BR3 reactor, was not soluble (non-mimas) by an industrial reprocessing unit
  - The products of the reprocessing were very difficult to recycle on an industrial level due to the large variety in isotopic composition of the uranium and plutonium

3.4. Request for quotation

The first study was the basis to set up a request for quotation. Three competitors responded to this with each a proposal for dry storage :

- ATEA-FRAMATOME (France) proposed the **NUHOMS** concept
- The Joint venture CFE-GNB Essen (Germany) proposed the **CASTOR 11 ST**
- Belgatom-Transnubel (Belgium) proposed the **TN24s**

- **Nuhoms (Atea-Framatome)** The main characteristics of this concept are :
  - The fuel is stored in a welded canister (figure 7)
  - One canister is provided with two welded lids, so no monitoring is necessary during the storage period
  - One canister can hold up to 75 assemblies, therefore three canisters would be loaded
  - The canister is transported in a lead container to its final storage

- The final storage is performed in concrete modules (figure 8)

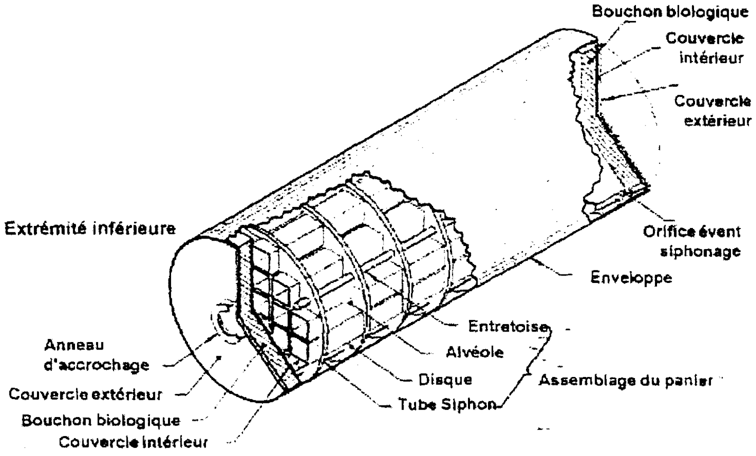

Figure 7. Welded canister to store the fuel

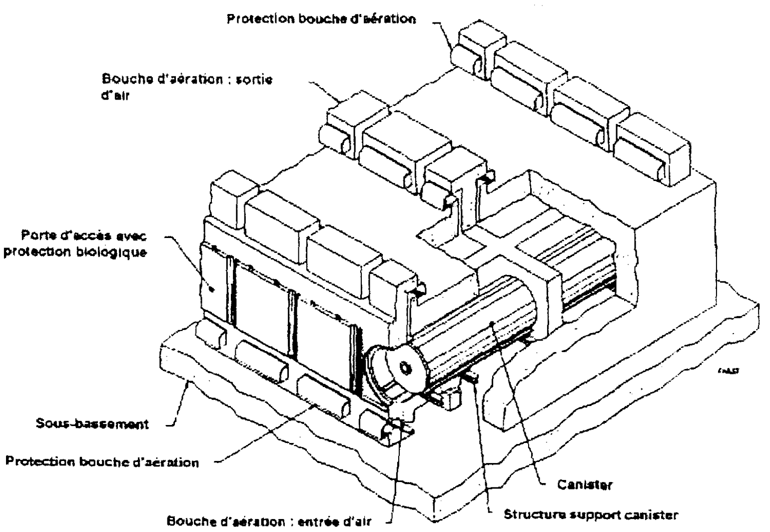

Figure 8. Concrete modules for final storage

- **TN 24S (Belgatom-Transnubel)** The main characteristics of this concept are :
  - The TN24S container is a transport package type B(U)F, with the following dimensions : height 4.2 m, diam. 2.5 m, weight 85 ton.
  - It is also used as a storage container.
  - The proposal implied 1 container to load all the assemblies. The loading is performed with a transport container, which can hold one assembly.
  - The container is closed with 1 lid and 2 metallic seals (figure 9). The inter-seal space is continuously monitored during storage.

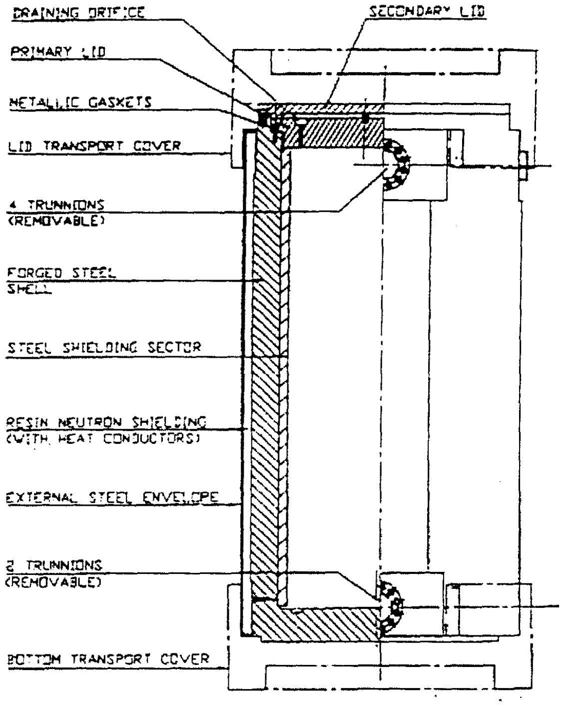

Figure 9. TN24s during transport

- **CASTOR 11ST (GNB Germany)** The main characteristics of this concept are :
  - The CASTOR 11ST container is a transport package type B(U)F, with the following dimensions : height diam. 1.4 m, weight 29 ton.
  - It is also used as a storage container.
  - One container can hold 30 BR3 fuel assemblies, therefore seven containers are loaded and one container is foreseen.
  - A container is closed with a double lid system and metallic seals. The inter-lid space is conti monitored during storage.

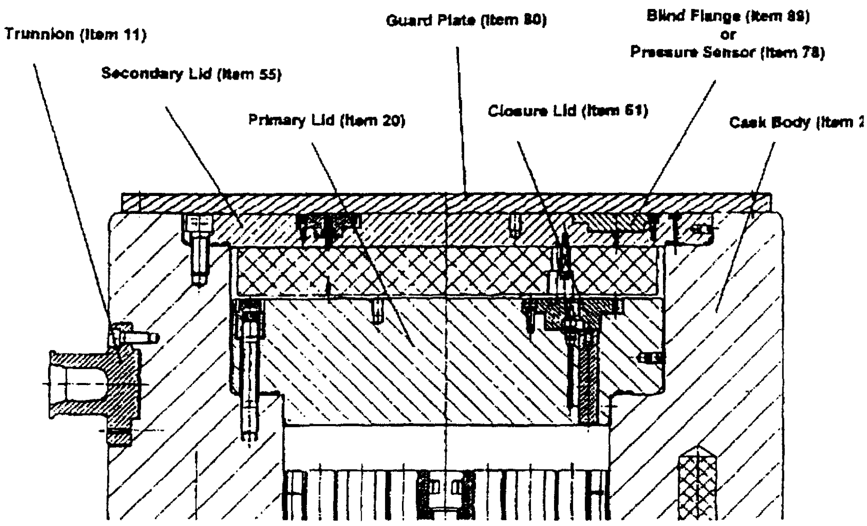

Figure 10. Castor 11ST : cover

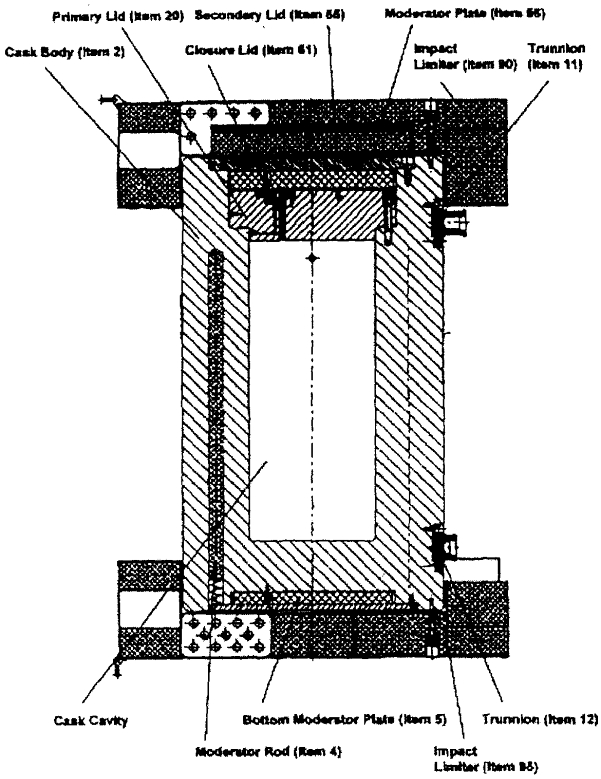

Figure 11. Castor 11ST : overview

- Storage of the container is performed in a dedicated building.

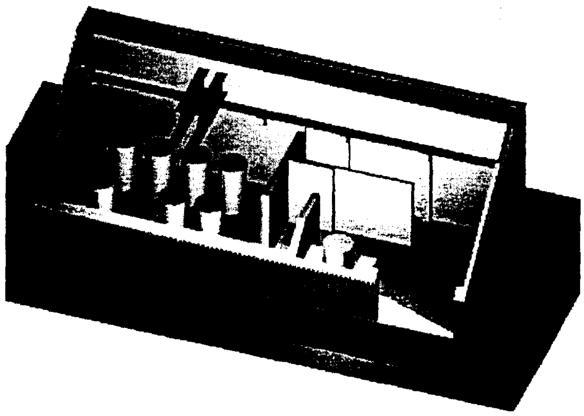

Figure 12. Drawing of the storage building (Belgoprocess)

3.5. Selection criteria

The following selection criteria were used to select one of the proposals :

- Long- and short term safety
- Licensing feasibility
- Technical feasibility

- Costs
- Possibility and easiness of intervention in case of incidents
- Retrieval of the fuel (feasibility, easiness, costs,...)
- Re-use of the container for final disposal
- Completeness of the quotation and solution description
- International references

➔ Finally the option proposed by the JV CFE-GNB was selected.

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4. CASTOR BR3®

4.1. The CASTOR BR3®

The CASTOR BR3® is derived from the CASTOR THTR and is designed as a transport and storage cask for accommodating 30 BR3 spent fuel assemblies. As a type B(U) cask fitted with shock absorbers, it meets the transport requirements according to the IAEA regulations and fulfils the conditions for cask storage in Germany. For the transport, the cask needs a Type B(U) licence from the Belgian competent authorities (FANC: Federal agency for nuclear control).

The necessary modifications to the original THTR design refer to an additional neutron shielding in the cask wall, the secondary lid and the cask bottom as well as to the basket. The basket is designed with closed loading tubes and can be handled separately by a crane even in loaded condition. This allows to unload a cask if some leak would be detected.

The main cask components are :

- The cask body constructed out of ductile cast iron foreseen of an internal cavity, which is coated to avoid corrosion. The dimensions of the cask (see table 2) are derived from the necessary shielding capability.
For neutron shielding, 30 axial boreholes are machined in the cask wall, which are filled with polyethylene rods.
- The primary lid, foreseen with an inner metallic and outer elastomer seal, closes the cavity and acts as first independent safety barrier both for the leaktight confinement of the fuel and for the radiological shielding. The closure of the metallic seal is controlled by a Helium leak tightness test performed on the inter-seal volume.
The primary lid contains an opening into the cavity, which is used for dewatering, drying and gas filling of the cask cavity after loading (He at 500 hPa abs).
- The secondary lid, also foreseen with an inner metallic and outer elastomer seal, serves as an additional shielding as well as a second independent tightness barrier.
An opening into the secondary lid, giving access to the space between primary and secondary lid, allows adjusting the pressure in the inter-lid space (He at 6000 hPa abs).
Additionally a pressure sensor system is mounted in the secondary lid to monitor the pressure in the inter-lid space hence providing a continuous check-up of the leak-tightness of the CASTOR cask during the long-term storage period.
- The inner basket consists of 30 hexagon tubes, which are mounted on a bottom steel plate. The tubes are made of borated stainless steel, the bottom plate of stainless steel and is provided with some small holes (diameter approx. 20 mm) for dewatering purposes. Apart from these small holes in the bottom plate, the basket structure is closed and designed for being handled in loaded condition.
The hexagons can accept both BR3 fuel assemblies and canisters loaded with fuel remnants.

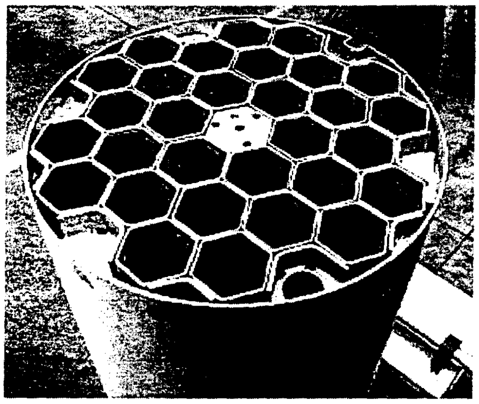

Figure 13. Innerbasket

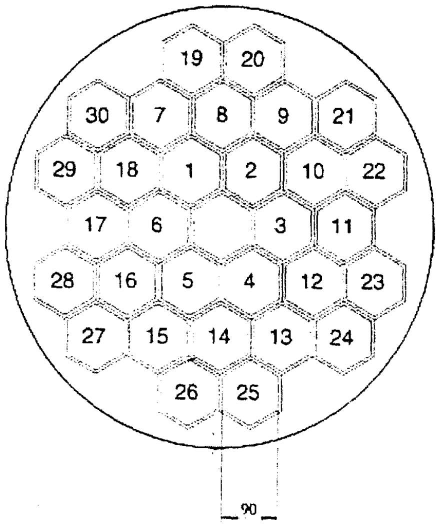

Figure 14. Drawing of the innerbasket showing dimensions and bundles positions

Part	Height (mm)	Ø (mm)	Wall thickness (mm)	Weight (tonne)
Cask body	2493	1428	370	20
Cask cavity	1652	690	-	-
Primary lid	270	911	-	1.3
Secondary lid	70	1122	-	0.6
Basket	1638	685	-	1
Inventory	150	(Hex) 85	-	0.8

Table 2. Dimensions and weights of the CASTOR BR3®

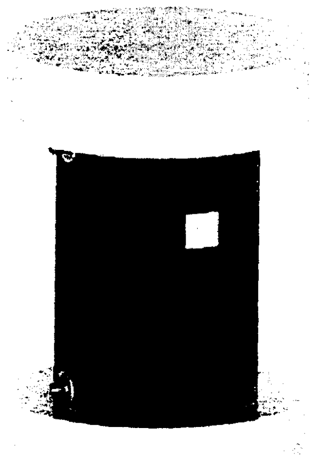

Figure 15. Transport Configuration

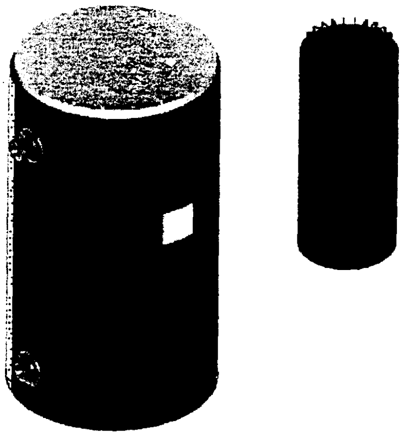

Figure 16. Storage Configuration

4.2. Parameters of the spent fuel

Due to the diversity of the spent fuel and in order to simplify the safety calculations, 4 fuel cases were defined which covered the most important and worst case of loading. The main characteristics of these fuel cases (residual heat production, neutron and gamma radiation) were used to perform the safety calculations (except for criticality) of the CASTOR BR3®. These four fuel cases are covering each assembly and canister, which will be evacuated in a CASTOR BR3®.

Calculations have showed that the worst case for criticality was a non-irradiated type Z fuel assembly. A CASTOR BR3® loaded with 30 fresh type Z assembly showed a Keff< 0.82 (Keff: effective neutron multiplication factor) after a type B(U) test.

4.3. Safety calculations

To obtain the necessary transport licence, a complete safety report has been issued by GNB (the manufacturer of the CASTOR BR3®). The following main items have been checked according to the IAEA regulations for the safe transport of radioactive material (1996 Edition No ST-1) :

- Study of the sealing system and activity retention of the Castor BR3®.
- Criticality calculations
- Shielding calculations
- Mechanical analyses under accident conditions
- Thermal analyses

For the storage licence, additional safety calculations had to be performed like :

- Cask behaviour in case of an earthquake
- Cask behaviour in case of a shock wave (caused by a nearby explosion)
- Mechanical interaction of casks in the storage facility
- Airplane crash onto the lid system (F16 fighter)
- Drop of the cask in the storage facility
- Fire
- Burial

4.4. Physical tests

On the CASTOR BR3® were no physical tests performed, the requirements for the safety analyses were demonstrated by extrapolation of test results of other types of CASTOR containers. These test results were obtained during physical test in the past in co-operation with the German authorities.
The most important tests were :

- 9 meter drop on the top, bottom, side wall and cask edge
- 1 meter drop onto a steel pin with the lid side and side wall
- 200 meter water immersion test
- Thermal test 30 min. at 800°C

4.5. Animation

(.avi film : 15 MB)

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5. Spent fuel preparation

5.1. Overview

During the consecutive core loadings, a large number of rods were withdrawn from their original assembly position to perform post-irradiation examinations or another irradiation in a different assembly position. This led to a situation that 25% of the assemblies were not complete and about 380 rods were stored in temporary storage cans.

5.2. Consolidation of fuel assemblies

Therefore, it was necessary to organise large consolidation campaigns in order to complete these fuel assemblies. The consolidation is quite time consuming. In total there were three consolidation campaigns over a total period of 12 weeks. Some internal transports, with a TN-6 container, were carried out because several rods were stored at the BR2 site.

Figure 17. TN-6 internal transport container

Figure 18. Installation for the consolidation of fuel rods

5.3. Leak tightness test "Wet Sipping"

Each assembly, which is completed and ready for evacuation, is submitted to a leak tightness test using the "Wet Sipping" method. The "Wet Sipping" is used in several nuclear power plants and is described in an IAEA document : "*Guidebook on Non-destructive Examination of Water Reactor Fuel (ref. Technical reports Series N° 322)*"

The principle of "Wet Sipping" is to detect fission products in the heated water wich circulates over the rods. Due to the higher temperature there is a pressure rise created in the fuel rods. A defective fuel rod will emit fission products, at least one of them (^{137}Cs) can be detected by γ-spectrometry.

Different cycles of the Wet-Sipping procedure :

A. Determination of the background

This step assures the operator that the empty sipping bottle contains no residual contamination. The bottle is rinsed with non-contaminated demineralised water during half an hour. After the rinsing a sample is taken and analysed by γ-spectrometry.

B. Determination of the blanco

This step assures the operator that all contamination wich has entered the sipping bottle during the loading of the fuel assembly is removed. The bottle (with fuel assembly) is rinsed with non-contaminated demineralised water during half an hour. After the rinsing a sample (blanco-sample) is taken and analysed by γ-spectrometry.

C. Circulation of heated water in a closed circuit

In closed circuit, heated water is circulating for two hours over the fuel assembly. Afterwards, a sample (sipping-sample) is taken from the water and analysed by γ-spectrometry.

D. Evaluation of the results

The sippingfactor (SF) is calculated as follows :
- SF = Value of the sipping-sample / Value of the blanco-sample

The decision if a fuel assembly is defective, suspected or leaktight is taken depending on the value of the sippingfactor :
- $SF < 2$: The fuel assembly is leaktight.
- $2 \le SF < 3$: The fuel assembly is suspected, the Wet Sipping has to be repeated.
- $3 \le SF$: The fuel assembly is defective.

During the "Wet sipping" of the assemblies, we found 11 suspected fuel assemblies of which all of them are already successfully treated to remove the non-intact fuel pins. An external safety inspectorate verifies the results of the leak tightness test.

Figure 19. Wet Sipping installations : pumps & loops

Figure 20. Wet Sipping installations : bottles in the racks

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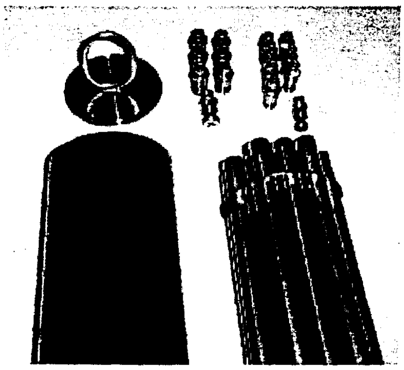
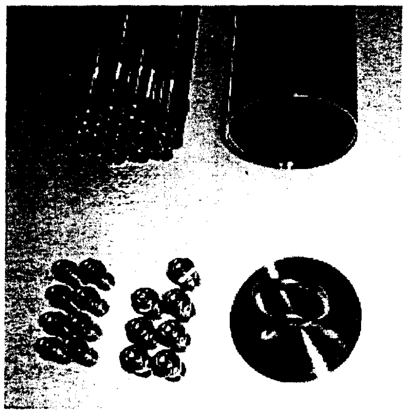
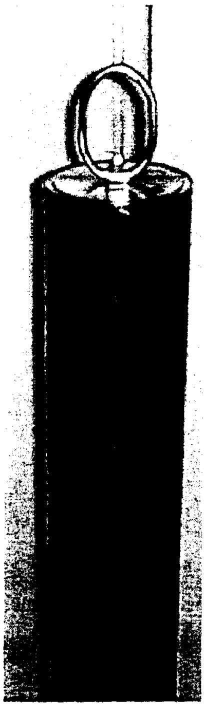
5. Spent fuel preparation

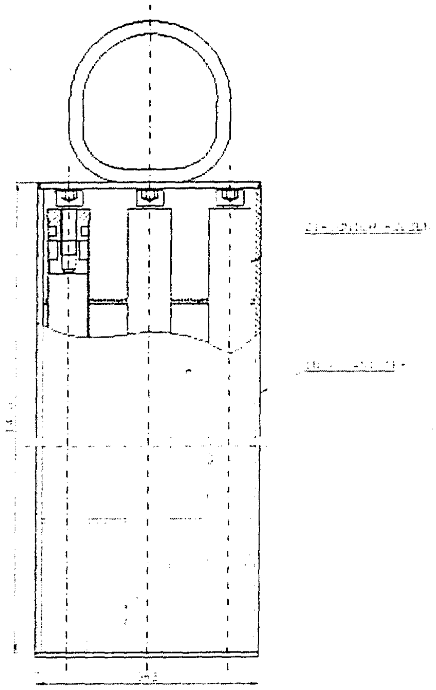
5.4. Canisters

Non-intact and cut fuel pins were loaded into canisters, which were welded and tested on leak tightness. Each canister can contain 15 fuel pins; pieces of rods were first loaded in a zircaloy tube before being put into the canister. The loading of the canister was performed in a hot cell facility at LHMA (see picture) and BR2.









Figures 21 to 25. Canister BR3

Figure 26. LHMA (SCK·CEN) facility

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6. Progress of the project

6.1. Production of the CASTOR BR3®

The production of the eight CASTOR casks started in March 2000. During production there was a quality assurance follow up according to the IAEA-regulations. Additionally SCK·CEN contracted AIB-Vinçotte to perform a spot-check follow-up of the CASTOR BR3® fabrication as a representative of the client.

The main production steps are :

- Casting of the cask
- Basket fabrication
- Fabrication of the lids and trunnions
- Assembly and testing

Figure 27. Casting of the Castor BR3®

Casting of the first CASTOR BR3®: 32 ton of liquid material was casted into a mould in a period of 2 minutes. The temperature of the material was about 1300° C.

Pictures of other parts :

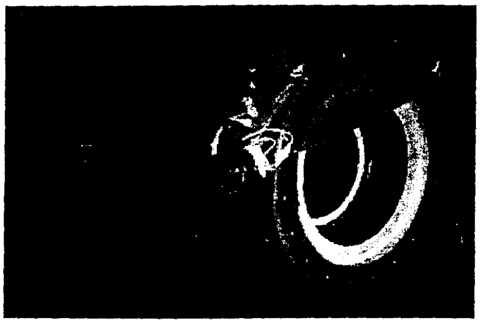


Figure 28. The cask body (painted and coated on the inner side)

Figure 29. The primary lid

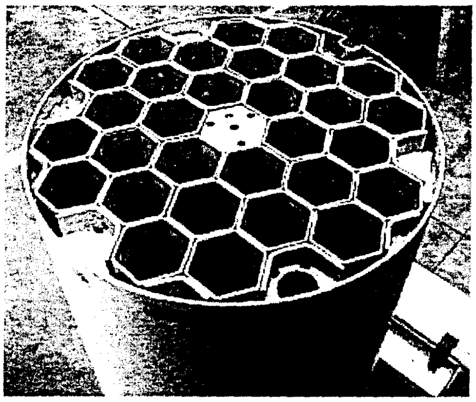


Figure 30. The basket

6.2. Transport licence

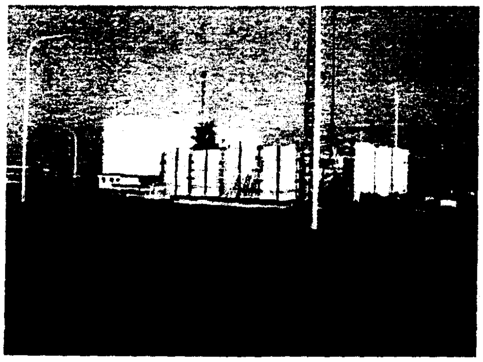
The study of the transport licence file is performed by AVN by order of the Belgian authorities and started in January 2001. The final Type B(U)F-96 licence is expected June 2002.

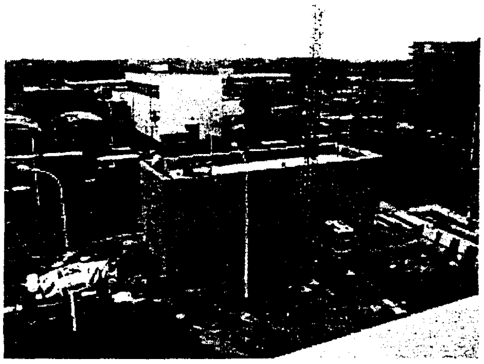
6.3. Construction of the storage building

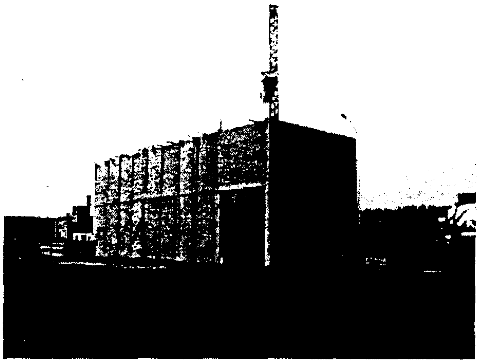
The construction of the building is started in June 2001 and was finished at the beginning of January 2002.

Pictures of the storage building :

1. Start of the construction

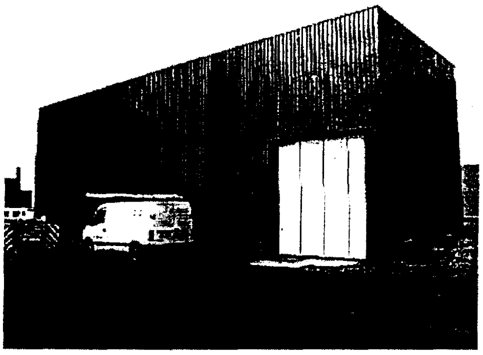






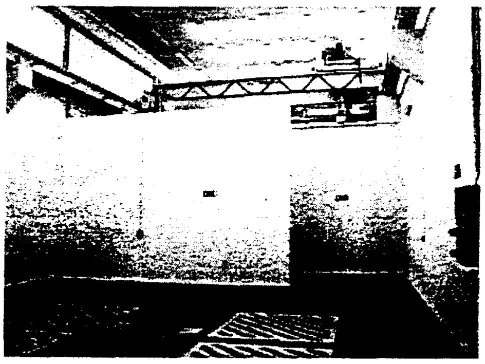
Figures 31 to 34. Start of the construction

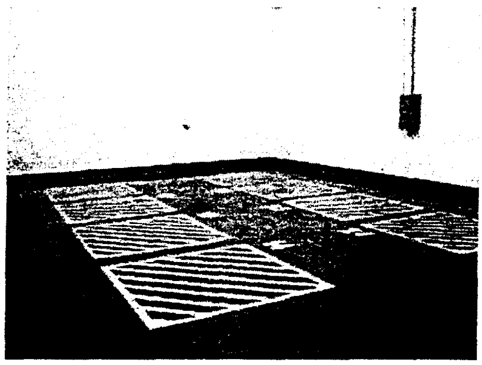
2. Finished storage building & project team



Figures 35 & 36. Finished storage building & project team

3. Inside the storage building





Figures 37 & 38. Inside the storage building

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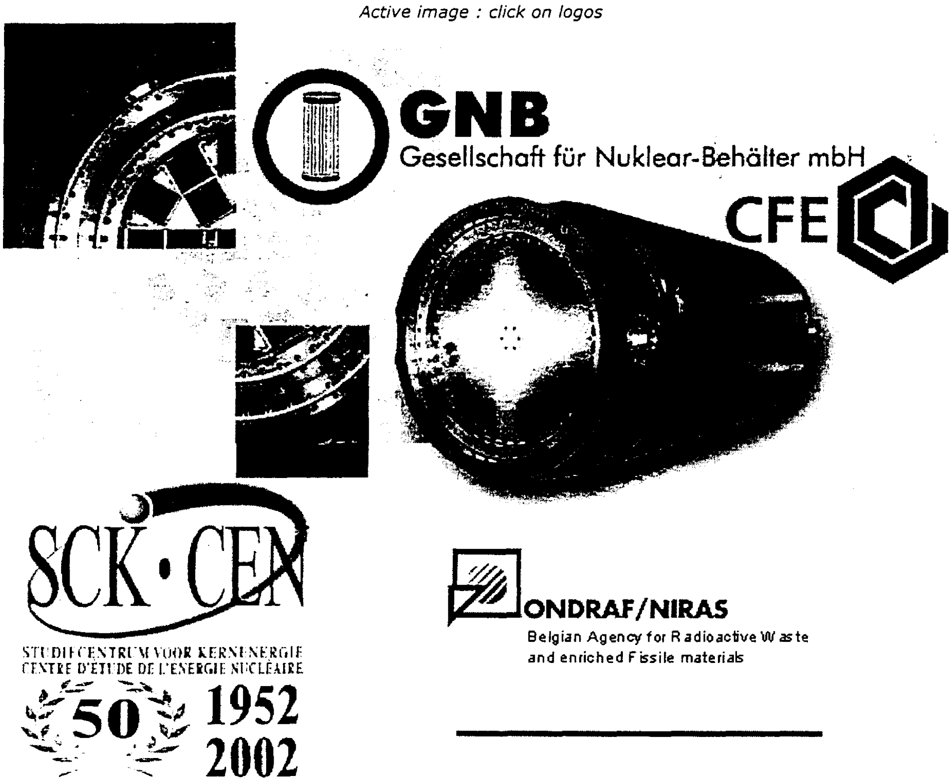
Back

Evacuation of the BR3 spent fuel

1. General information on the fuel
2. Existing Storage facility
3. Description of the project
4. CASTOR BR3®
5. Spent fuel preparation
6. Progress of the project
7. Contact people

Active image : click on logos



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行政院所屬各機關因公出國人員出國報告書

(出國類別：實習)

赴比利時SCK-CEN研究所實習核設施除役技術

(附件四 – 附件六)

服務機關：行政院原子能委員會核能研究所

出 國 人：職　　稱：薦任助理研究員

　　　　　姓　　名：郭鴻達

出國地區：比利時

出國期間：91年11月2日至91年12月7日

報告日期：92年2月6日

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附件四

BR3 計畫產出之投稿論文：4 份

- Decontamination Strategy for the Dismantling of Strongly Contaminated Loops: the Practical Case of the Dismantling of the BR3 PWR Auxiliary and Primary Loops

- Overview of Recycling Technologies for Decommissioned Materials: Lessons Learned during the Dismantling of a Reactor

- Nuclear and Non-Nuclear Safety Aspects in Nuclear Facilities Dismantling: The Example of a PWR Pilot Decommissioning Project

- Dismantling of the BR3 Reactor Pressure Vessel

3rd European Forum of "Radioprotectique"
Radioprotection and logic of dismantling
October 2 – 4, 2002
La Grande Motte, France

Decontamination strategy for the dismantling of strongly contaminated loops : the practical case of the dismantling of the BR3 PWR auxiliary and primary loops

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Aknowledgements : Mathieu Ponnet, Olivier Emond, Yves Demeulemeester, Fernand Vermeersch, Jérôme Dadoumont, Vincent Massaut

ABSTRACT

In a PWR reactor, the primary and auxiliary loops are generally strongly contaminated with ^{60}Co so that the dose rate will be a radioprotection issue during their dismantling. At SCK•CEN, for the dismantling of the BR3 reactor, we decided, for radioprotection reasons, to perform as soon as possible after the reactor shutdown, a full decontamination of the loops with the objective to reduce significantly the dose rate. After the operation, we successively created a new evacuation way for the dismantled materials, removed the contaminated insulation, dismantled the loops and finally dismantled the large pieces of equipments (still underway).
For ALARA study, we use now the VISIPLAN software, developed at SCK•CEN, which allows to predict the radiological exposures in a complex environment. In parallel with these dismantling operations, we decontaminated the dismantled pieces using different decontamination techniques (abrasives processes using a special ventilated booth and chemical processes using the Medoc installation) with the objective to reach the nuclear recycling criteria or the clearance values.
We will present the main lessons drawn from these operations on the safety point of view (radiological as well as classical safety) but also on the materials management point of view. The management of the dismantled materials is indeed a complex task and our objective is to minimize the radioactive waste volume and to minimize the overall materials management cost.
This paper is focused on the results of the management of the pieces arising from the dismantling of the most contaminated loops of the plant. We show that it is possible to recycle most of the metallic materials either in the nuclear world or in the industrial world by reaching the respective recycling or clearance criteria.

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The following describes in a simple way:

- The full decontamination of the primary loop and associated results and lessons drawn from it;
- Some operations needed to create a new evacuation way at BR3 for the material arising from the dismantling of the RPV and the primary containment;
- Overview of the dismantling techniques;
- Overview of the decontamination techniques and evacuation routes followed by the wastes produced;
- Some about the QA system for the material flow at BR3;
- Some conclusions.

INTRODUCTION

The BR3 (for Belgian Reactor n° 3) was the first PWR installed in Europe. In service since 1962, it was shutdown in 1987 after 25 years of operation and 11 campaigns. It is a quite small reactor with a net electrical power of 10.5 MW_e or a thermal power of 40.9 MW_{th}. At the end of its operating life, the European Commission, in the framework of its five-year plan of RTD selected in 1989, the BR3 as one of the four pilot projects on decommissioning of nuclear installations.

The main steps of the decommissioning programme up to now were:

- Full System Decontamination of the primary circuit in 1991 (the CORD® process),
- Dismantling of the high active thermal shield with three different techniques, of the Vulcain internals and of the first set of Westinghouse internals (30 years decay time) by mechanical cutting up to 1996,
- The dismantling of the auxiliary circuits started in 1995 and is still going on,
- The dismantling of the reactor pressure vessel ended in 2000 after its removal in one piece ("world première"),
- The construction and start of the exploitation of thorough decontamination processes for dismantled pieces in the period 1996 to 1999 (the MEDOC installation).

For the dismantling of the primary loop, we decided to reduce the dose rate by chemical decontamination, to dismantle the pieces and to sort them following their specific evacuation route, and finally to decontaminate the pieces with the objective to minimise the amount of materials disposed as radioactive waste.

FULL SYSTEM DECONTAMINATION

As part of the dismantling strategy for the BR3 reactor, a Full System chemical Decontamination of the reactor primary circuit was conducted in 1991. The objectives were to reduce the radiation dose rates in the vicinity of the low and non-activated components and to limit the transfer of surface contaminants during subsequent dismantling operations.

The Full System Decontamination has been performed using the CORD® process developed by Siemens AG KWU. The CORD® process (acronym for Chemical Oxiding-Reducing Decontamination) comprises 3 successive steps : an oxidation step with permanganic acid, a decontamination step with oxalic acid and a cleaning step with ion exchangers.

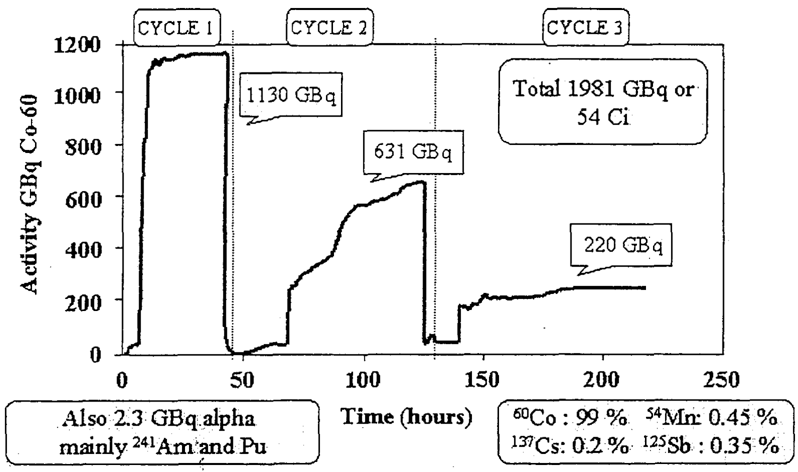


Figure 1. ^{60}Co Activity release during the 3 decontamination cycles
(ref. date 1987 i.e. 4 years after shutdown)

For the BR3 system, 3 successive decontamination cycles at an operating temperature of about 80°C to 100°C were performed. The operation itself lasted nine days of continuous operation. The following main systems were decontaminated : the reactor primary loop (fuel unloaded but internals loaded), the purification loop and a part of the Residual Heat Removal System (the Shutdown Cooling Heat Exchanger and the Emergency Shutdown Condenser).

The total volume of the loop is about 15 m^3 and the total surface treated about 1200 m^2; most of the surface, steam generator included, is of Stainless Steel 304.

During the 3 cycles, a total of 2 TBq of gamma emitters was released with ^{60}Co as the dominating γ nuclide, which is no less than 90% of the activity of the primary circuit (Ref.date 1991). A total quantity of about 2.3 GBq alpha activity was also removed from the circuits among which about 185 mg (equivalent to 629 MBq) of plutonium. A total quantity of about 33 kg of oxides or corrosion products was removed; this corresponds to a mean release of about 2.8 mg of oxide/cm^2. A mean Decontamination Factor close to **10** has been achieved, with a broad spread of individual values ranging from 0.1 (redeposition of activity in a horizontal pipe) to 31 (steam generator) according to the measurement location. In the plant container, the decontamination operation has deeply modified the picture, as far as working conditions are concerned: the ambient dose rate has been reduced by a factor about 10 and varied after the operation between 20 and 60 μSv/h. In the purification circuit, the ambient dose rate was then around 10 to 30 μSv/h.

Some concluding remarks and lessons can be drawn from the experience of a Full System Decontamination operation on a reactor in dismantling phase :

⇒ ***Process operation :***
The process applied is a smooth process, only a few and minor operational problems were encountered. This could only be achieved by a careful and detailed preparation of the operation. It requires a reactor in full satisfactory operational condition and experienced and qualified operators. For a dismantled reactor, this operation has to be performed maximum 4 to 5 years after shutdown to have all equipments still operational.

⇒ ***Operational problems :***
Some leakage's occurred during the operation due to corrosion of chromium plated pistons of the Charging and Circulating pumps; one set was replaced between 2 cycles to limit the leak. The leakages were recovered by the plant system.

⇒ ***Post-operation problem :***
A pollution of the reactor pool occurred during the unloading of the reactor internals resulting in a high turbidity; this pollution was due to a small amount of insoluble ferrous oxalate and loose crud still present in the circuits. This pollution could easily be removed by the plant filtration system.

⇒ ***Workload :***
The total workload was about 19,000 man.hours.

- the preparation itself took about 1 year and 75% of the workload;
- the operation itself only 2 weeks and 17% of the workload;
- the post-operation and waste evacuation took about 2 years and 8% of the workload. The evacuation of the High Active resins was the most delicate operation; the transport was performed in a special container and the conditioning was performed in a dedicated installation of the Belgian Nuclear Power Plant Doel.

⇒ ***Radioactive waste production*** :
A limited amount of secondary waste was produced (1.3 m^3 of ion exchange resins for a total treated surface of about 1200 m^2). Nevertheless, the quantity used was higher than foreseen which shows the difficulty to estimate the exact crud quantity for a complex system. This implied the unforeseen removal of active resins during the decontamination operation. Moreover, the conditioning of resins by cement embedding increases the primary waste volume by about a factor of 5 due to the limited capacity of this matrix (max 20 to 25%).

⇒ ***Radiological impact :***
An important dose reduction (DF~10) in the vicinity of the primary loop was achieved. The collective dose uptake for the performance of the operation amounted to 158 man.mSv. The major part (85%) of the dose was received during the preparation works, which took about one year. Under the operating deck, i.e. around the primary coolant loop, the ambient dose rate was reduced around 0.06 to about 0.08 mSv/h but at certain locations hot spots of several mSv/h were still present.

⇒ ***Impact on dismantling :***

→ Further impacts on the dismantling and on the evacuation of the dismantled pieces are experienced:
→ the general cleanliness facilitated the dismantling of the circuits in dry conditions (less contamination dispersion);
→ some low activated pieces were decategorized from MLW to less expensive LLW;
→ the dismantled less contaminated pieces will be easier to decontaminate to free release levels or to evacuate without treatment for melting for recycling;
→ the dose rate at the bottom of the refuelling pool has been reduced by a factor 2 to 3 and amounted after the operation to about 0.3 mSv/h; this facilitated the separation operations of the reactor pressure vessel.

To conclude, the operation was a success from the technological, radiological and economical aspects and such a decontamination operation should be promoted for future dismantling operations in nuclear reactors.
The figure hereafter shows the implementation of the decontamination devices in the existing installation (primary and purification loops).

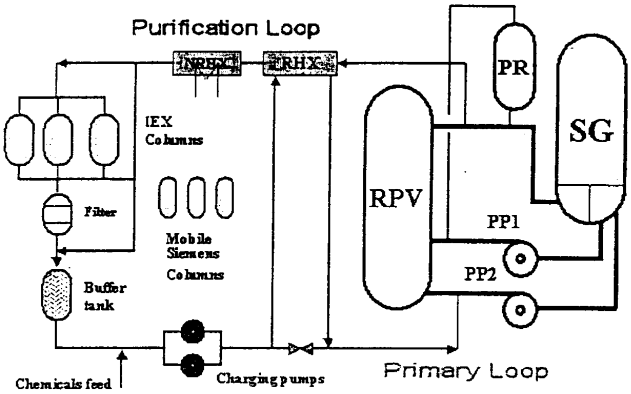


Figure 2. Flow sheet of the BR3 loop and the associated decontamination equipments

PARTIAL DISMANTLING OF SOME LOOPS AND CREATION OF A NEW EVACUATION WAY

The BR3 containment has only a personnel lock situated above the operating deck of the reactor. For the evacuation of the pieces, a new evacuation way had to be cut through the containment building and an adjacent room called the Siba room.
The sequence of operations was as follows:

- dismantling of some loops situated in the Siba room,
- cutting of a first opening in the concrete walls of the Siba room and of the containment,
- further dismantling and evacuation of loops in the vicinity of the concrete yard.

At that point, it was possible to perform the removal of the contaminated insulation around the piping and the vessels. This operation had to be done by a specialised firm because the insulation contained asbestos. The safety precautions taken were stronger than for a nuclear dismantling yard: the site was completely isolated from the rest of the building, the ventilation system was equipped with supplementary pre- and absolute filters, the personnel and materials locks were equipped with showers to avoid any transfer of asbestos fibres in the rest of the building. The removal of the contaminated insulation greatly improved the work environment around the loops. The contamination level in the air was very low allowing going in this zone without protective masks. The mask was only requested when dismantling works were performed.
Thereafter, the piping (free of insulation) situated in the vicinity of the concrete yard could be dismantled allowing the installation of the concrete dismantling machines (diamond cutting machines: saw or cable cutting). A specialised firm of the construction industry made the new evacuation way.
The asbestos removal yard accounts for about 46 % of the collective dose because of the long duration of the work (3000 hours), the number of operators (from 8 to 10) and the mean dose rate (between 0.02 and 0.04 mSv/h). Without decontamination, the mean dose rate would have been about 10 times higher so that the workers would have received an exposure higher than the monthly authorised value (2 mSv/h). We would then have been obliged to manage the dose uptake with a higher number of persons.

DISMANTLING WORKS IN THE PRIMARY CONTAINMENT

The primary containment contains the whole primary loop and the associated purification and safety loops. Before starting the dismantling of the large components (steam generator, pressuriser, primary pumps, large tanks...), it was necessary to remove all the piping with their associated auxiliaries (instrumentation, electric cables..). In total 20 loops were dismantled or adapted corresponding to a removal of about 40 tons of materials.

Due to the complexity of the environment, a detailed organisation had to be set up comprising:

- the preparation of a general strategy with the help of an ALARA planning software,
- modifications to a number of loops which have to be maintained for further use and need,
- dismantling of light accessible pieces to allow the installation of scaffoldings and a new transport rail,

- dismantling of the remaining loops and of the primary piping.

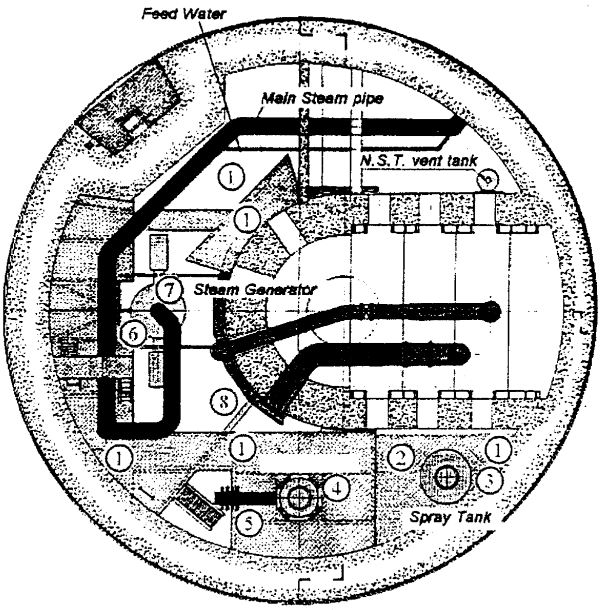
The general strategy developed considered the following aspects:

- The detailed elaboration of the different tasks; for each task, an estimation of the manpower, the technical issues, the classical and radiological hazards.
- The planning must fit into the general D&D planning. The evacuation of the active pieces from the dismantling of the RPV in a shielded container through the primary containment was the main conflicting task because it occurred in the middle of the D&D dismantling work.
- The optimisation of the D&D work to minimise the dose uptake. Due to the complex environment, we used a 3D ALARA simulation software developed at SCK•CEN, the so-called VISIPLAN software. This allowed us to estimate and compare the collective dose for different dismantling scenario and to optimise the work organisation. Comparison has also been made with a "manual" estimation.
- The size reduction of the dismantled pieces, the sorting and evacuation of the pieces according to their specific evacuation route is often the bottleneck in a dismantling work. We had to organise the work to avoid the accumulation of cut pieces at the outlet of the dismantling yard.

A detailed ALARA study was necessary and requested by the Health Physics department due to the foreseen length of the work (about 6 months), the number of operators involved (up to 4), and the dose rate (foreseen up to 11 mSv/h). A detailed survey was done revealing the presence of important hot spots (about 11 hot spots with dose rates from 0.1 to 11 mSv/h were identified). This is shown in figure 3.

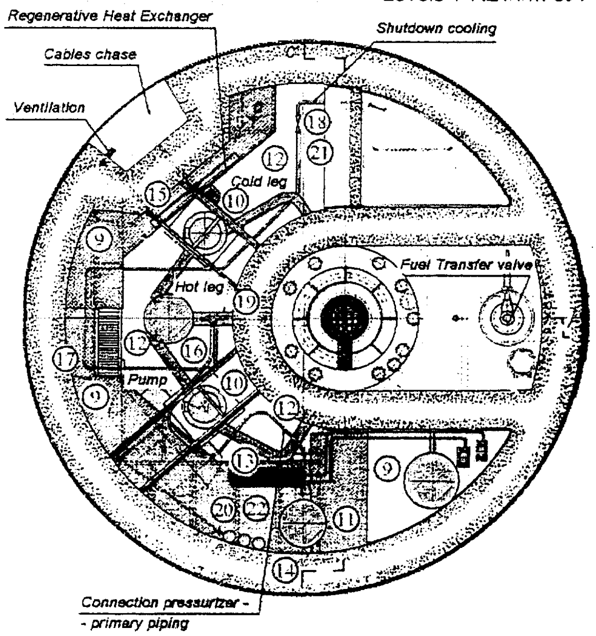
The Alara study confirmed that the removal of hot spots was beneficial for a reduction of the collective dose. Nevertheless, it was not always technically feasible. We first removed the "easiest" hot spots at the ground level and then we removed progressively the hot spots starting at the + 7 m level down to the –4.8 m level.

Levels + 0.208 m et – 4.805 m



Point n°	Description	Dose rate (µSv/h)
1	Ambient dose rate	1 – 3
2	Contact Spray Tank	5 – 10
3	SP valve	50
4	Contact Pressurizer	7 – 9000
5	Pressurizer Discharge pipe	900
6	Contact Steam Generator	1 – 15
7	Ambience around SG	8 – 30
8	Mixed Bed Tank	20

Levels + 7.214 m et + 3.353 m



Point n°	Description	Dose rate (µSv/h)
9	Ambience foot bridge	6 – 40
10	Top motor Primary Pump Main Coolant	120
11	Contact DDT tank level 0	40
12	Contact Primary pipes	15
13	Spray Heat Exchangers	50 – 100
14	Coolant Sampling Heat Exchangers (Herpi's)	500 – 1200
15	Hot spot MC Pump 2	1000
16	Elbow Shutdown Cooling pond side	3500
17	Elbow SC containment wall side	1100
18	SC valve	>700
19	Ambience max 1 m above ground level	30 – 70
20	HDT + DDT tanks	1200 – 2500
21	Hot spot bottom containment on floor	1300
22	Leak Off Tank	11000

Figure 3. Radiological mapping before the dismantling

The dismantling operations were performed in a sequence taking into account the Alara aspects and the classical safety aspects. The sequence selected is given hereafter:

- Preparation phase
  - Installation of a confinement for the whole yard and of a ventilation system with pre- and absolute filters.
  - Installation of a cutting workshop at the ground level containment outlet; this zone was mainly equipped with a band saw machine.
  - Clear identification of the piping by markers to facilitate the materials follow up (facilitating the sorting).
  - Modification of some vital loops : compressed air, treated water, and neutron shield tank loop (cooling with chromated water).
- Dismantling of some hot spots at the ground level : coolant sampling, heat exchangers, drain collectors, ...
- Cutting and evacuation of piping around several equipments, electric cables and their supports, small tanks, evacuation of the rotors of the primary pumps, in order to allow the placement of scaffoldings and a transport rail with pneumatic cranes, evacuation of the main steam and feed water piping using an oxyacetylene burner.
- Systematic dismantling of the remaining
- General water cleaning of the yard to allow the removal of the loose contamination and the further working without masks.
- Cutting of the primary pipes: in-situ with a lathe rotating cutter for the segmentation in pieces of several m long, size reduced to max 1 m long with a band saw in a cutting stand at the containment outlet.

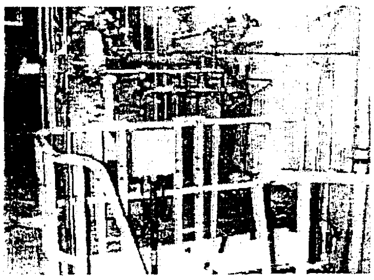
The collective dose, total workload and material dismantled for these operations are respectively :
11.95 man.mSv - 4124 man.hours - ~ 40 t.

Up to now, for the whole dismantling works realised under the operating deck, the total collective dose amounts only to 49.85 man.mSv ; besides what mentioned above, it includes :

- Dismantling Regenerative heat exchanger
- Removal of loops in the Siba room and the cutting of the metal sheet of the containment
- First opening in the Siba room and removal of piping there
- Removal of the contaminated asbestos
- Dismantling of Safety Injection loop and other piping
- Final cutting of the concrete for the new evacuation way
- Partial Removal of several loops
- Size reduction in cutting workshop
- First cutting of the primary pipes at the reactor bioshield outlet
- Dismantling component cooling, pressuriser surge line, primary pumps

Without decontamination, the exposure would have been much higher. The installation of shielding's or the remote dismantling in such a confined environment would have been very difficult so that we can assume that the exposure would have been about 10 times higher i.e. about 500 man.mSv.

Figure 4. shows as example, the situation around the pressuriser before and after the operation, with the ventilation shaft.



OVERVIEW OF THE DISMANTLING TECHNIQUES

In general, preference was given to mechanical cutting techniques such as portable band saw, reciprocating saw, lathe pipe cutter ... because they produce no aerosol or dust, only chips or iron filings, easily caught. Nevertheless, sometimes we had to use thermal techniques such as the oxyacetylene torch or dust producing techniques such as grinding, that needed to work in a controlled and ventilated area with air monitoring.
A first size reduction stand was installed at the yard outlet. We used there mainly a medium capacity band saw (max 300 mm width capacity) or the large band saw (max 500 mm capacity) which had previously been used for the underwater cutting of the reactor internals : this was especially used for cutting in pieces the primary pipes.
Further size reduction was done inside a ventilated cutting workshop situated in the auxiliary building. Inside this booth, we can use thermal cutting, grinders or circular saw. The booth is equipped with a strong ventilation system with regenerable prefilters followed by absolute filters.

SORTING OF THE PIECES

The next step in the dismantling strategy is the sorting of the materials according to their specific evacuation route. As our main objective is the minimisation of radioactive wastes, we are using two main evacuation routes:

- ***recycling route*** i.e. reuse of the materials inside the nuclear world.
- ***clearance route*** i.e. unconditional reuse of the materials for the industrial world.

The crucial point in this process is ***the sorting of the materials.*** Specifications are established to help the operator in its choice. The sorting of the materials leads to the creation of batches i.e. materials, which follow the same evacuation route. An interactive database has been constructed to allow the follow up of the materials from the creation of the batch up to its final evacuation (see further : QA system for the material flow at BR3).

TREATMENT OF VERY LOW OR NON RADIOACTIVE MATERIALS BY MELTING

Nowadays, "nuclear" melting facilities are in operation in several countries for the treatment of very low level metallic wastes. To be cost effective, these installations must have a sufficient throughput. Up to now, Belgium does not have any available facility so that conditioning contracts were signed with facilities abroad.

Melting for recycling in the nuclear world

Low level radioactive materials are recycled in the nuclear world.. The melted materials are used for the fabrication of shield blocks or for the fabrication of radioactive waste containers. SCK•CEN had an agreement with GTS-Duratek in the USA; the recycled materials were used as shielding for the DOE facilities. Unfortunately, at this moment, the melting activities of GTS-Duratek are postponed for an undetermined period of time due to loss of clients.

Melting for Clearance

Some dismantled materials are either very low contaminated, very difficult to measure or not homogeneously contaminated. For these materials, it can be advantageous to send them to a nuclear foundry. Melting offers several advantages:

⇒ It decontaminates the metals by volatilization of some nuclides (e.g. ^{137}Cs) or by transfer to the slag (e.g. heavy nuclides such as alpha emitters).
⇒ It allows an accurate determination of the radionuclides content thanks to the homogeneity of the metal melt.
⇒ The amount of secondary waste (dust, slag) is rather low.

We have an agreement with the Studsvik facility in Sweden. A first transport of 18 tons was done in 2000.

DECONTAMINATION TECHNIQUES

For metals, we use mainly :

- ***Manual washing or cleaning in an ultrasonic rinsing bath***: mainly for pieces only slightly contaminated on the surface by deposition of contamination on external surfaces (demineralized water piping, structural pieces, instrumentation boxes..). For the US cleaning, we use a rinsing tank of 2 m^3 capacity equipped with 9 US transducers.

- ***Manual Polishing with metal grinding and polishing machines***: hand held machines manually operated inside a ventilated booth.
  The technique was applied on the main steam pipes, which after dismantling were very slightly contaminated, localized on the top part of the external side of the pipes (the inside was clean). The contamination level was between 1 to 2 Bq/cm^2 $\beta\gamma$ and was due to deposition of contamination during the dismantling works. The polishing technique was then applied in a ventilated booth situated in a low background area (the former machine hall of the plant) to enable a direct measurement of the piece after treatment. This operation was a total success as 100 % of the pieces were cleared totally respecting the surface contamination criteria ($\beta\gamma$<0.4 Bq/cm^2).

- ***Stripping of electric cables*** to separate the contaminated insulation from the copper. The electric cables can be sent to the Tecnubel Company who operates a semi-automatic stripping machine.

- ***Abrasives decontamination***: mainly used for rusted or painted pieces of simple geometry in which the contamination is fixed in the oxide layer or in the paint (structural equipment, beams..). An installation called ZOE is used at BR3 for the treatment of pieces up to 3 t and 3 m long maximum. This installation uses a wet abrasive process operated manually inside a ventilated booth. For smaller pieces, we can send our pieces to an automatic dry sand blasting unit operated by Belgoprocess; however, the pieces must fit inside a 200 l drum.

- ***Hard chemical decontamination with the MEDOC Cerium process***: mainly used for stainless steel pieces heavily contaminated up to 20,000 Bq/cm^2 ^{60}Co (primary loop, tanks,...). The Medoc installation has a capacity of about 20 m^2/batch, which corresponds, to 0.5 to 1 t of metals. Typically, one treatment takes one day.
  Moreover, the Medoc installation has been recently used for the successful decontamination of the Steam Generator and in the shortcoming also for the pressuriser. More results will be available after the cutting of these two equipments allowing then more accurate measurements.

Figure 5. Wet sandblasting installation

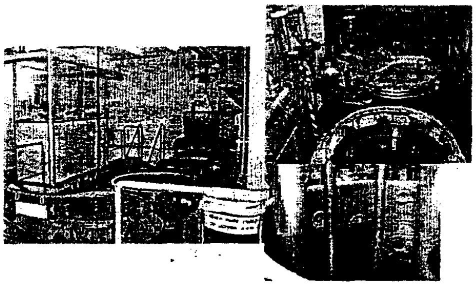


Figure 6. the Medoc installation

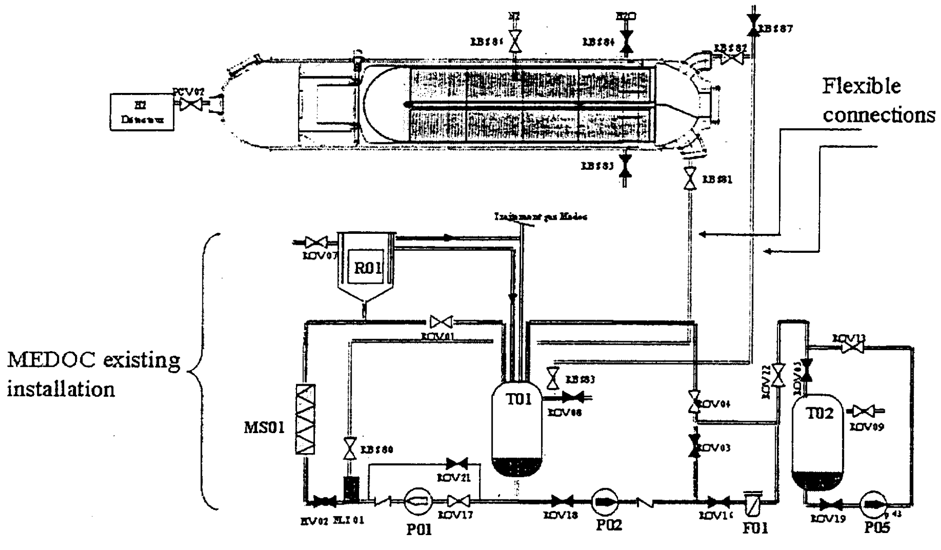


Figure 7. Decontamination of the Steam Generator using the Medoc installation

Results obtained for the treatment of the materials arising from the dismantling operation

The dismantling works performed so far have produced about 40 tons of metals. We have already sorted 33 tons of materials between their respective evacuation routes. The primary pumps with their casings represent 7 tons; they still must be further dismantled.

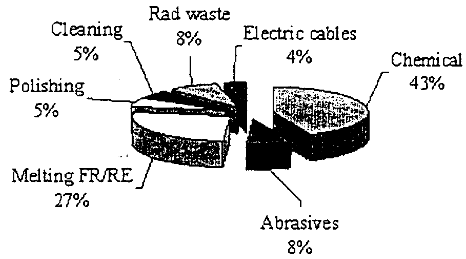


Figure 8. Dismantling of the contaminated loops situated in the containment building.
Distribution of the 30 t of materials through the different evacuation routes

Some remarks on this distribution:

- The chemical decontamination unit will treat a major part of the materials.
- The melting route for recycling or for free release will be selected after a further sorting and characterization of the pieces to cope with the respective specifications of the melting firms.
- The abrasives process will mainly be used at BR3 for the large pieces and at Belgoprocess for the smaller ones.
- The quantity of radioactive waste produced is remarkably low (about 8% , i.e.~ 3.2 tons).

The contamination level of the dismantled pieces was very variable: it varied from several thousands of Bq/cm^2 for the primary pipes to some Bq/cm^2 for the main steam pipes. All the pieces situated inside the containment were contaminated to a certain degree due to the primary steam leaks, which occurred during the reactor exploitation period.

The main lessons drawn up to now from these decontamination operations are:

- the ***polishing technique*** was a total success as 100 % of the pieces were cleared totally respecting the surface contamination criteria ($\beta\gamma$<0.4 Bq/cm^2).

The ***chemical decontamination*** of all the stainless steel pipes and pieces and in particular of the primary pipes was also quite successful. The decontamination campaign of 10 t is finished and about 80 % of the treated mass can be cleared either by direct measurement (for the primary pipes) of the surface or by measurement of the massic contamination level for the complex pieces. The rest will be sent to the melting for free release route. Some pieces still present some hot spots difficult to remove or some pieces are slightly activated. We discovered that the primary pipes sections (about 1 t) situated in front of the biological shielding openings were slightly activated (activation level 0.5 to 1 Bq/g) which prevents their direct clearance.

The ***abrasives decontamination*** of the structural pieces is running since 1996. We know by similar experience that from 80 to 90 % of the treated mass can be cleared after treatment.

QA SYSTEM FOR THE MATERIAL FLOW AT BR3

When dismantling a nuclear installation, a complex material flow is generated due to the great variety of materials and their possible destination. Moreover, a small part of the material is contaminated and activated implying severe safety measures. The dismantler has to meet different requirements. Due to the nuclear nature of the materials, several organisms request their traceability and, when the decontaminated material is going to be recycled, the requirements are even much severe.

To meet these requirements, it is necessary to collect all relevant information during the dismantling. To do it in a structured way, a QA system can be implemented. In the case of the BR3 project, such a QA system was started in the beginning of 1998, based on the ISO9002 norm. This implies that all operations have to be written down in procedures : they guarantee that the correct information is collected and that the operations are carried out in the same way, having thus an automatic control.

The system is at this moment almost ready to be proposed for accreditation.

Requirements

When dismantling a nuclear facility, besides the owner of the plant (the SCK•CEN in the case of BR3), other organisms are controlling the dismantling activities. They concern mainly the safety authorities, national (FANC) as well as at the SCK•CEN level, and the National Organism for Radioactive Waste and Fissile Materials (ONDRAF/NIRAS) as manager of the technical liabilities, funding which represents the main source of financement of the dismantling project.
The dismantling team together with the above mentioned organisms have defined their requirements concerning the generated material flow. These requirements concern the traceability (administratively as well as physically), the final destination of the material and the free release measurements or characterization of the material.
The traceability aims at knowing at any moment the origin and destination of the dismantled material. Throughout the process, it is thus possible to quantify and to manage the different material flows (radioactive waste, free released waste, ...). The dismantler himself wants to know, in order to manage the material flow, where and how much demolished material is located at the site. This information is necessary to obtain a clear view of the project's progress and of the possible needs for storage or handling.
All the dismantled material will go to a final destination (waste, melting, scrap, ...), each with its specific requirements. In case of waste, the material has to filful the national waste requirements set out by ONDRAF/NIRAS. When the material is for unconditional recycling (scrap), it has to meet the severe free release limits.
In case of other final destination, the material has to satisfy a variety of specific requirements.

Needs

To meet the above mentioned requirements, the following needs have to be fulfilled :
- to get all the relevant information about the dismantled materials ;
- to avoid loss of necessary information ;
- to standardize the dismantling approach so that everyone knows his job ;
- to describe the different treatment methods to be sure that the people use them correctly ;
- to describe and validate the different characterization and contamination measurements to be sure that the read out of the measurement is the reality and that the destination requirements are met ;
- to have an organizational description of the selected work so that there is a clear overview of the process ;
- to prove to the public that everything we do is within the national and international legal framework.

Solution

To meet the needs deriving from all the above mentioned requirements, the rules have to be clearly defined and the organization must be very efficient. All these rules have been brought together in a quality assurance system. As dismantling can be considered as the production of a product (material destinated for reuse or nuclear waste), the BR3 dismantling team has chosen the ISO9002 QA system.

When the management of BR3 decides to dismantle one specific part of the plant following the decommissioning plan, the steps could be defined as follow and are shown on figure 6 :

- The procedures with regard to the preparation phase have to give as much as possible information about the part to be dismantled and the way to follow as well as the proposed final destination. Therefore, sampling, characterization and history are very important (the latter one especially for the release of material).
- The procedures give primarily information about the safe execution of the work and the gathering of information (dose uptake, volume of dismantled material, ...). The reports are important as first step on the way to the administrative traceability.
- The dismantling finishes with the sorting of the material in batches : the sorting criteria are of great importance. The unique batch number directly linked to the computerized database allows indeed an easy physical traceability.
- The material flow procedure is very important because here the complete organization of data collection and transfer of the batches are described. The collection, receipt, centralization and processing of the information are described in detail in order to meet the requirements (our own-developed database plays a key role).
- The next step after the different treatment methods is the characterization or free release measurement of the batches. The free release procedure, issued by the Health Physics Department, together with the procedures giving the different measure- en characterization methods, are the main documents.

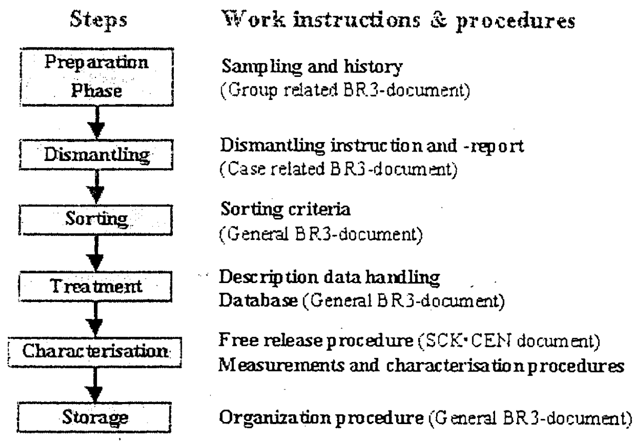


Figure 9. Steps and procedures for the organization and follow up of dismantled material at BR3

Return of experience

Although the QA system is still being implemented, it is inherent to a QA system that the administrative tasks increase. However, while reducing the amount of supplementary work to a minimum, one has to keep in mind that it is indispensable in order to keep a good traceability. Also, the reluctance from some operators can partly be explained by the fact that the older ones are less familiar with the philosophy of a QA system and are often driven by ancient habits of doing. One could reduce this reluctance by showing the benefits of the system, thus increasing people's motivation. However, in the case of BR3 it is very difficult to translate this easily into quantitative information.

A positive aspect is that, by defining the working methods, the goals are reached in a very efficient way. The system gives also an incentive to a more structured way of working thus increasing the clarity of work organization and work methods for the operators. By giving details, possible difficulties in working methods and organization and sometimes unknown before (especially internal communication) can be detected and remediated.

As QA implementation is quite recent, its progress is mainly going on by distributing the different procedures and work instructions. Internal audits have shown some weaknesses which are taken into account and solved in order to ask an external audit for accreditation.
Moreover, measurement and characterization methods have been finalized and validated, especially with a view to unrestricted release of dismantled materials. This action is carried out in close collaboration with other departments of SCK•CEN, i.e. health physics and departments specialized in nuclear measurements. This collaboration is one of the challenges for the future implementation of the QA system.
Nevertheless, up to now, the part of the system already implemented, i.e. the data base and batch identificator system, has proven to be really efficient and thus gives positive indication for the finalization of the full QA system implementation.

CONCLUSIONS

The strategy followed: "full system decontamination – dismantling – segmentation and sorting – thorough decontamination " allows to

⇒ Minimise the workers exposure, working then following the ALARA principle, let's even say the ASARA for "as safe as",
⇒ Minimise the amount of radioactive materials disposed as radioactive waste
⇒ Optimise the amount of materials either recycled in the nuclear world or directly recycled in the industry.

This strategy is economically feasible and saves natural resources.

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OVERVIEW OF RECYCLING TECHNOLOGIES FOR DECOMMISSIONED MATERIALS: LESSONS LEARNED DURING THE DISMANTLING OF A REACTOR

INTERNATIONAL ATOMIC ENERGY AGENCY
International Conference on Management of radioactive waste from non-power applications: Sharing the experience.
Malta, 5-9 November 2001

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Ref: 052/01-16

OVERVIEW OF RECYCLING TECHNOLOGIES FOR DECOMMISSIONED MATERIALS: LESSONS LEARNED DURING THE DISMANTLING OF A SMALL PWR REACTOR

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Abstract

The dismantling of the BR3 research reactor produces quite large masses of contaminated materials, mainly metals or concrete. The main management routes are: conditioning of the radioactive wastes and disposal, recycling of radioactive materials in the nuclear sector and the recycling of cleared materials in the industrial sector or their evacuation as industrial waste.

The paper gives an overview of the main management routes followed with their associated decontamination techniques. During the dismantling of a complex system, the sorting of the materials according to the selected evacuation route is the critical point. The sorted materials are classified as homogeneous "batches" and their follow up is guaranteed up to the final evacuation.

The main processes beside the evacuation as radioactive waste are:

⇒ Melting in a nuclear foundry either for clearance or for reuse. As Belgium does not have any "nuclear" melting facility, contracts were signed with companies abroad.

⇒ We study the recycling of slightly radioactive concrete for the preparation of a "radioactive mortar" for the conditioning of metallic radioactive waste. The R&D work is presented in this paper.

⇒ Several decontamination processes are used for metals and for concrete with the objective of unconditional clearance of the materials.

  ⇒ Abrasive processes and aggressive chemical processes are mainly used for the treatment of metallic pieces.

  ⇒ Scabbling or shaving is mainly used for the treatment of concrete.

For unconditional clearance, the issues related to the guiding values, the measurement procedures and the measurement techniques are dealt with and the experience obtained during the dismantling of our facility is presented.

1. INTRODUCTION

BR3 is a small pilot 10 MWe PWR used for operator training and fuel testing, shutdown in 1987 after 25 years of operation. It was selected as one of the four pilot projects of the European Commission for its R&D program on Decommissioning of nuclear installations. The decommissioning project started in 1989. In 1991, a Full System Decontamination of the primary loop reduced the dose rate in the vicinity of the primary loop by a factor 10. The same year, a first high active internal, the 5.4 t thermal shield was dismantled underwater by 3 different dismantling techniques, the EDM cutting, the milling cutter and the plasma arc torch. Mechanical cutting, essentially milling cutter and band saw, were selected for the further dismantling of the two sets of internals; the original "Westinghouse" internals (33 years decay) and the "Vulcain" internals (7 years decay).

The 28 t Reactor Pressure Vessel has been dismantled in 1999-2000 using mainly a circular saw for the horizontal cutting in rings and a vertical band saw for the cutting into segments. Dismantling of contaminated circuits including the primary loop is also performed using

mostly hands-on and semi-automatic cutting techniques. The next important step is the dismantling of heavy active pieces (reactor pressure vessel cover and bottom, neutron shield tank) and large contaminated components (steam generator, pressurizer). For these pieces, High Pressure Water Jet Cutting with abrasives using a remote operated arm will be used. [1] [2] [3].
Minimizing the amount of radioactive waste and recycling or clearance of the dismantled materials have always been our main objectives. The paper is focused on the main issues and results related to the management of the dismantled contaminated materials.

2. EVACUATION ROUTES

Dismantling of a nuclear facility produces quite large quantities of materials and associated gaseous, liquid and solid effluents. Not only primary materials are produced i.e. the items dismantled but also secondary materials like tools, equipment, new hardware for dismantling and decontamination and secondary effluents and swarfs from the dismantling operations. [4] [5].

The major solid materials coming from the dismantling operations are:

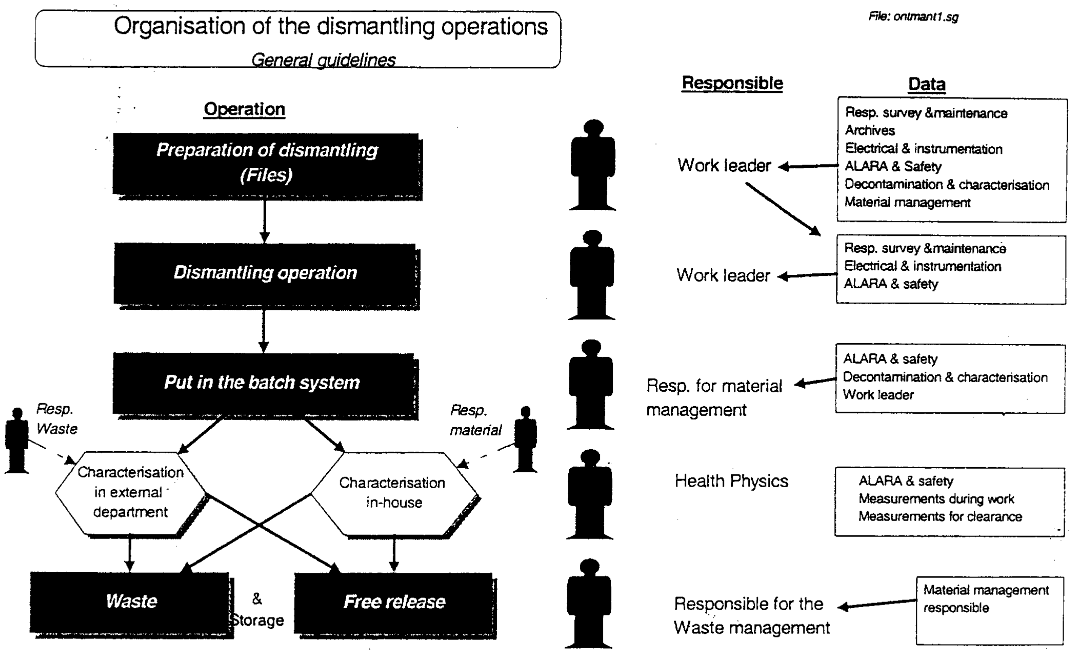
- Burnable wastes such as protective clothing, wood from ventilated hoods, laboratory furniture...
- Low to High level massive metallic wastes such as reactor internals, reactor pressure vessel, primary pumps, reservoirs, valves, and structural materials...
- Low to High level super-compressible metallic wastes from the same sources as above plus e.g. electric cables, light supports, contaminated instrumentation...
- Massive concrete pieces from slightly activated or contaminated slabs, floors, shielding walls, room walls...
- Concrete, bricks and super-compressible rubble from demolition activities of activated or contaminated materials.
- Sludges from deposits in reservoirs and liquid sumps.
- Various lightweight non-metallic super-compressible materials such as thermal insulation.
- Special waste such as contaminated lead bricks and shielding.

Three main material categories can be distinguished:

- Material which can be directly considered as conventional and treated as such, e.g. evacuated as industrial waste or recycled in the industry: alternator, tertiary loop, equipments outside the controlled area.
- Material which has to be evacuated as radioactive waste, e.g. activated materials or strongly contaminated material that cannot be technically or economically decontaminated or cannot be recycled or re-used: reactor pressure vessel and its internals, highly activated concrete, contaminated insulation materials....
- Material which, a priori, has to be considered as radioactive material, but as alternative to its evacuation as radioactive waste can be cleared unconditionally after decontamination, cleared after melting or recycled in the nuclear industry: contaminated piping, reservoirs, pumps, structural equipments, contaminated concrete...

3. WORK ORGANIZATION

The dismantling of a nuclear facility is a complex task. Therefore the dismantling operations are divided in hundreds of different tasks. For each task, a working procedure is established. This procedure gives the details of the work to be done and makes an analysis of the safety aspects (conventional and radiological). The work is only started after approval from the Health Physics.



The main steps followed for a typical dismantling work such as the cutting of a contaminated loop are:

- On site ***dismantling*** in large pieces *e.g. cutting with a reciprocating saw of pipes.*
- ***Cutting*** in small pieces in a ventilated workshop *e.g. cutting with a plasma arch torch of a 3 m long pipe into 4 pieces (allowing for clearance measurement device, decontamination and radwaste packaging in 400-l drums).*
- ***Sorting*** *e.g. separation between pipes, pumps and electric motors etc.*
- ***Identification*** *e.g. this pipe is put in a batch for stainless steel and is contaminated at less than 1000 Bq/cm^2 in ^{60}Co.*
- ***Temporary storage***: *e.g. the cut pipes are stored in a 300-l container in a storage rack in the auxiliary building.*
- ***Treatment***: *e.g. a batch is treated by chemical decontamination.*
- ***Characterization***: *e.g. radiochemical characterization is performed using hand held β monitors.*
- ***Evacuation***: *e.g. the pieces are sent as scrap materials.*

In this process, the crucial point is the ***sorting***. It has to be carried out as soon as possible after dismantling (cutting) the piece in order to guarantee the traceability i.e. where does it come from, what is its history? The sorting of the material must be well prepared in advance to accelerate the operation. The operator must know the destination of the material: is it foreseen to be evacuated as radwaste, sent to melting for recycling, sent to the chemical or to the physical decontamination unit...? Specifications are established to help the operator in its choice but it is not always an easy task because the pieces from the same origin can go to different routes. For example, a contaminated pump can be sent either to the chemical decontamination or to melting for clearance or recycling or be evacuated as radioactive waste. The decision depends on the contamination level, the geometry of the pump, the materials composition, the nature of the contamination...

The sorting of the materials leads to the creation of "batches".

A batch is a group of materials that will follow the same evacuation route. A batch can be a 300-l plastic container, a 200-l drum, a 400-l drum or an individual large piece e.g. a reservoir or a heat exchanger.
Every batch carries a unique identification label corresponding with one record in a computerized database. The content of a batch, its status and its location is known therewith at each moment.

All relevant information is collected:

- A unique identification number is written on a label fixed on the batch; this label gives the content of the batch, its weight and the evacuation route selected.
- The actual status (e.g. buffer storage, characterization, route selected, evacuated,...) is reported in the database.
- Finally, a document is edited with all the necessary approvals. In function of the selected route it will be a clearance document, a request for treatment as radioactive waste or an authorization to send to a melting facility.

All this information is put into a "user friendly" database. This database can only be modified by one responsible person and accessed in "read only" by a few persons.

4. CONDITIONING PROCESSES FOR RADIOACTIVE WASTE STREAMS

The radioactive wastes from the D&D operations at BR3 are conditioned mainly by BELGOPROCESS, a subsidiary of ONDRAF/NIRAS [6].
The solid wastes are conditioned essentially in a new facility, called the CILVA installation, which comprises:

⇒ an incineration facility of a capacity of 10 t/week solids and 1 to 3 m^3/week burnable liquids

⇒ a pretreatment facility, in which the solid wastes are sorted, cut and eventually pre-compacted at 140 t.

⇒ a super-compaction facility for 200-l drums with a 2000 t hydraulic press of 6000 drums/a capacity.

⇒ A conditioning unit for immobilization and embedding: A cement matrix is used to fill 400-l drums in which supercompressible pellets or non compressible wastes are stacked. This installation also includes an active mixer for embedding of wet wastes like ion exchange resins and sludge.

⇒ After solidification, inspection and measurements, the drums are transferred in an intermediate storage building.

5. TREATMENT OF RADIOACTIVE METALS BY MELTING

Nowadays, "nuclear" melting facilities are in operation in several countries for the treatment of low level metallic wastes. To be cost effective, these installations must have a sufficient throughput. Up to now, Belgium does not have any available facility so that conditioning contracts were signed with facilities abroad.

5.1. Melting for recycling in the nuclear world

Low level radioactive materials are recycled in the nuclear world. The melted materials are used for the fabrication of shield blocks or for the fabrication of radioactive waste containers. SCK•CEN has an agreement with GTS-Duratek in the USA; the recycled materials are used as shielding for the DOE facilities. The materials must respect composition and radiochemical criteria. The secondary wastes are conditioned and disposed off by Duratek. Up to now, we have sent, in agreement with all the competent authorities, 26 t of mild and stainless steel arising from the dismantling of very low contaminated or activated pieces. The specific mean activity of the pieces sent lied around 25 Bq/g for ^{60}Co and 6 Bq/g for ^{137}Cs.

Future transports are being considered for materials produced during the dismantling operations.

5.2. Melting for Clearance

Some dismantled materials are either very low contaminated, very difficult to measure or not homogeneously contaminated. For these materials, it can be advantageous to send them to a nuclear foundry for free releasing them after melting. Melting offers several advantages:

- It decontaminates the metals by volatilization of some nuclides (e.g. ^{137}Cs) or by transfer to the slag (e.g. heavy nuclides such as alpha emitters).
- It allows an accurate determination of the radionuclides content thanks to the homogeneity of the metal melt.
- The amount of secondary waste (dust, slag) is rather low.

This practice has already been used in Belgium for dismantled waste. SCK•CEN has performed a first melting campaign in Studsvik Radwaste Sweden. This first transport of very low activity materials comprised secondary reheaters with copper tubes, a carbon steel massive plinth and a variety of Carbon Steel and Stainless Steel small pieces stored in 200- and 400-l drums. About 18 t with an average activity of 0.26 Bq/g of ^{137}Cs and 0.15 Bq/g of ^{60}Co have been melted in September 2000.

All the ingots (17.2 t) produced could be unconditionally cleared. The secondary waste (slag and dust) contains all the Cesium activity; it represents a mass of 1.1 t or a volume of 0.9 m^3 i.e. 5% of the initial waste volume. The secondary waste is sent back to Belgium for conditioning whereas the metal ingots are cleared in Sweden.

Future transports are being considered for materials produced during the dismantling operations.

The materials will be separated by type (carbon steel, stainless steel, copper, aluminum); lead and galvanized steel are not accepted in this foundry. The paint must be removed from the pieces either by sand blasting in our facility or by sand blasting in the Studsvik Radwaste facility. The presence of organic matter and encapsulated water must also be avoided.

6. RECYCLING TECHNIQUE FOR LIGHT RADIOACTIVE CONCRETE

The dismantling of the activated bioshield around the Reactor Pressure Vessel and in the refuelling pool will lead to the production of about 650 t of slightly activated concrete. The activation products present are mainly ^{133}Ba, $^{152,154}Eu$ and ^{60}Co with activation levels lower than 100 Bq/g. A R&D programme has been started with a specialized center in the building industry (CSTC/WTCB, Scientific Centre for the Building Industry) to study the possibility to reuse the concrete as raw materials for the conditioning of radioactive wastes [7].

Instead of considering this huge amount of material as a radioactive waste to be conditioned either as super-compressible waste or as standard waste, an alternative is to try to re-use it as raw materials for the production of "radioactive " cement grout. This cement grout can then be used for the conditioning of heterogeneous radioactive waste. It can be used for the conditioning of the compacted disks arising from the super-compaction of light metallic wastes or it can be used for the conditioning of massive metallic pieces.
In the super-compaction process, the compacted disks, arising from the compression of 200-l drums, are piled inside a 400-l drum and the voids between the disks and the drums are filled with cement grout. In the direct conditioning process, the cement grout is directly poured inside the 400-l drum which contains the massive pieces placed in a basket.

The cement grout used for the conditioning of radioactive waste must comply with stringent specifications among which the compression strength and the workability are the most important. The grout is composed of a mixture of cement, sand, water and a fluidifier. In the case of the reuse of radioactive concrete, the idea is to replace the sand by radioactive concrete. The radioactive concrete must therefore be crushed and sieved to deliver a product which can cope with the specifications of the grout.

The study comprises as main steps:
- Crushing and sieving of inactive heavy concrete
- Selection of an optimized grout formula (Cement/Water/Fluidifier)
- Qualification tests of the grout.
- Crushing and sieving tests of the active heavy concrete
- Active qualification tests with the formulas selected with the inactive tests.

The tests were performed in a pilot installation of the KEMA(NL) company. The installation comprises a jaw crusher and a sieve unit. The installation was adjusted to optimize the fines production.
By an adequate choice of the grout composition, it was rather easy to respect the two main contradictory specifications:
- a fluid grout which will adequately fills the voids in the waste package;
- a high compression strength after 90 days (> 20 N/mm²).

Finally, active tests were done in collaboration with Belgoprocess which, in Belgium, is responsible for the radioactive waste conditioning.
For example, a mixture composed of ¼ of dry cement grout and ¾ of recycled heavy concrete was prepared and tests performed on the final product. The workability was satisfactorily and the compression strength reached 49 N/mm² after 28 days.

In conclusion, it was demonstrated that it is technically possible to re-use radioactive heavy concrete as raw materials for the preparation of "radioactive grout" respecting the stringent specifications of the waste conditioner.

7. DECONTAMINATION TECHNIQUES

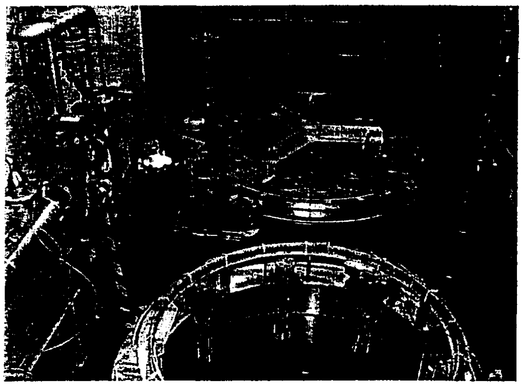
The steadily increase of the conditioning and disposal costs as well as environmental concern and public perception are pushing the nuclear sector to decrease the amount of radioactive waste and hence is a strong incentive to the development of thorough decontamination processes and procedures for the clearance of obsolete radioactive suspected materials and their reuse in the industrial sector or their evacuation as industrial waste.

For metals, we use mainly:

- ***Manual washing*** or cleaning in an ultrasonic rinsing bath: mainly for pieces only slightly contaminated on the surface by deposition of contamination on external surfaces (demineralized water piping, structural pieces, instrumentation boxes...).
- ***Dry or Wet abrasive decontamination***: mainly used for rusted or painted pieces of simple geometry in which the contamination is fixed in the oxide layer or in the paint . A wet installation called ZOE is used for the treatment of pieces up to 3 t and 3 m long maximum (heavy support beams externally contaminated). We send also materials to Belgoprocess which treats pieces smaller than 80 cm and lighter than 25 kg in an automatic dry blasting unit.
- ***Polishing and grinding***: mainly used locally for pieces which present some residual contamination before or after another treatment. As typical example, we treated the main steam pipe which was slightly contaminated on the external upper part by radioactive dust deposition during the dismantling operations.
- ***Miscellaneous techniques*** such CO_2 blasting, electric cable stripping... can be performed by specialized firms.
- ***Hard chemical decontamination with the MEDOC Cerium process [8]***: mainly used for stainless steel pieces heavily contaminated up to 20,000 Bq/cm^2 ^{60}Co (primary loop, tanks,...). The Medoc process is based on the use of a strong oxidant Ce^{4+} which dissolves at 80°C the oxide layer and attacks the base metal as well. The reactant is regenerated continuously by reaction with ozone gas. The Medoc installation (Picture 1.) has a capacity of about 0.5 to 1 t of metals per batch. Up to now, we have processed 21 t of stainless steel and we have started the decontamination of carbon steel pieces using a simplified Medoc process. It is worthwhile to note that we prepare the in-situ decontamination of our stainless steel steam generator.

Up to now, about 80 tons of metals have been treated in these different decontamination workshops. About 10 to 20% were not directly cleared; they are then sent to a nuclear melting facility for further decontamination and clearance or for recycling in the nuclear industry; the choice between the melting facilities being a function of the residual contamination.

Picture 1: Medoc installation

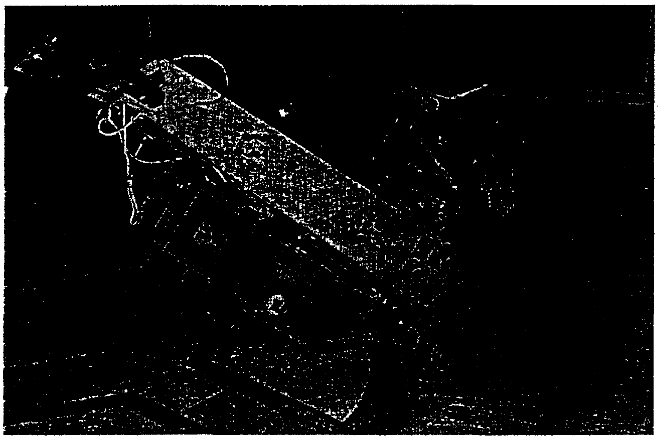


For concrete, we use mainly:

⇒ Scabbling with various types of hand held or floor scabblers, (Picture 2)
⇒ Shavers with diamond tools,
⇒ Jack hammers, hand held or remote operated such as the Brokk.

Most of the concrete decontamination works are performed by a specialized firm Tecnubel. Up to now, about 200 tons have been cleared and recycled in the construction industry.

Picture 2. Floor scabbler for concrete decontamination



8. CLEARANCE OF METALLIC MATERIALS

It is not only necessary to use efficient decontamination techniques, it is also imperative to proove that the materials are radioactively clean.

The clearance of radioactive materials requires a combination of factors to be successful:

- Procedures and well-defined clearance criteria: a consensus is not yet achieved on international level and generally a case by case management is still applied. IAEA, EU, OECD are progressively converging towards some harmonization. The competent authority in Belgium has recently established clearance levels. In our case, the Health Physics department under supervision of the competent authority establishes the specific procedures. These procedures are then applied for the clearance of materials from the BR3 dismantling.
- A strict follow-up of the dismantled materials comprising origin of the materials, treatment performed and characterization results.
- The traceability of the materials must be guaranteed at each step: this can only be achieved with a strong Quality Assurance program, presently being implemented.

The characterization of materials to be cleared is still a difficult topic. Materials candidate for clearance without melting can be subdivided into 3 categories:

- Materials of simple geometry for which a 100% surface measurement is possible using hand held β monitors. For these materials, surface specific clearance values and the procedures are well known. The values used are 0.4 Bq/cm^2 for $\beta\gamma$ emitters and 0.04 Bq/cm^2 for α emitters.
- Homogeneous materials such as concrete rubble for which only volume or mass measurement is possible. For these materials, mass specific values are followed and measurement procedures are available (e.g. γ spectrometry of the whole amount in a 200-l drum or statistical sampling after homogenization).
- Pieces of complex geometry and/or heterogeneous materials (pipes internally contaminated, pumps, valves.): the question is how to prove that the activity level is lower than the current clearance values? A procedure, based on a double measurement method, has been worked out.

The procedures followed are:

- Hand held monitors for easy to measure materials; 100% of the surface measured twice at a max 3 months interval for materials submitted to a decontamination treatment (sweeping effect).
- For homogeneous materials, we actually use the Q2 Canberra spectrometer with HPGe detectors for measurements of 200-l drums.
- For heterogeneous materials, we have two possibilities:
  - Either the materials are sent to a nuclear foundry, which allows a further decontamination and a reliable measurement thanks to the homogenization.
  - Or we combine two measurements techniques:
    - A gross gamma counting with scintillation detectors for measurements of individual pieces or of small batches (1/10 of a 200-l drum). The apparatus is the ESM CCM monitor FHT 3035.
    - A Q2 spectrometer for the determination of the specific activity per individual gamma nuclide.
- For large pieces of equipments we have used the ISOCS system equipped with a HPGe detector.

The Q2 spectrometer is well known whereas the ESM gross gamma counter for the small batches as well as the Isocs system were recently used and the results obtained are further detailed.

8.1. The ESM CCM gross gamma counter

The ESM is used to carry out 'selection' measurements, whereas the Q^2 spectrometer performs clearance measurements.

As the Q^2 spectrometer could not guarantee that the measured activity would not be concentrated on one item (or on a small volume), an extra measurement was requested to search for 'hot spot'. As the geometry of the items to be measured could be complex, and as the contamination could be located in an 'unreachable' area, a hand held measurement was considered as useless. Therefore a 4 π geometry was considered to be a better solution, the response of the device being less dependent on the geometry. We selected the Cobalt Coincidence Monitor (CCM) developed by ESM: the FHT 3035.
This monitor has the following characteristics:

- 4 π plastic scintillation chambers with CCM technology,
- closed chamber (cube of 600mm side length) with 6 plastic detectors of 500 x 500 x 50 mm^3,
- 5 cm lead shielding on all the faces.

The purpose of the ESM is to:

1. Get closer to the assumption of an homogeneous drum, for the Q^2 measurement
2. Chase 'hot spot'
3. Assort between 'candidate for clearance' and 'not candidate', to increase the probability of clearance by the Q^2 measurement

Pictures 3 and 4: ESM-FHT 3035 installed in a low background area

The purpose of the measurement in the ESM is to avoid the presence of hot spots in a drum. If one of the alarm level is exceeded, the small batch is refused and the operator can make a manual check to further detect the presence of a hot spot and eliminate it.

The batches measured by the ESM are collected in a 200-l drum, which is then measured by the Q2 spectrometer for the clearance measurement. The experience has shown that there is a good correspondance between the ESM and the Q2 techniques and that indeed "hot spots" can be detected and removed from the batch.

8.2. Measurement of large pieces

During the dismantling of the secondary loop of the BR3, we detected some localized contamination on massive pieces:

- Detection of very low level ^{137}Cs contamination on the tubes of the secondary steam reheaters (Smears test, tube sampling). The heat exchangers (2 t each, 4.5 m long, 1 m diameter) were sent to a smelting facility for a successful clearance.
- Detection of ^{137}Cs contamination on the turbine shell localized on an accessible surface. After decontamination by wet abrasive, this 3.6 ton part could be measured directly and cleared.
- Detection of low level ^{137}Cs contamination on the rotor of the turbine. Qualitative measurements were done with a portable NaI(Tl) detector which allowed to detect the presence of contamination and to perform localized decontamination treatments (chemicals decontamination, wet abrasives, CO_2 decontamination). However, we missed a qualified measurement method, which could be accepted by the Health Physics for the clearance measurement. The ISOCS was then selected for this heavy piece of equipment weighing 6 t.

Pictures 5and 6 show the turbine rotor during the final survey with the Isocs. It could be cleared unconditionally; all the measurements were below the 0.4 Bq/cm^2 criteria.

The ISOCS allowed to detect some local contamination on these large objects and in conditions where conventional contamination measurements are difficult and sometimes non-representative. Another advantage for the ISOCS method is the relatively short calculation time (minute range) and the possibility to use different available mathematical models.

9. CONCLUSIONS

The management of dismantling materials, with the objective of minimization of the amount of radioactive waste by applying decontamination and clearance or recycling, is a complex task due to the high variety of materials, the high variety of contamination levels and the low level measurement issues.

Up to now, we have been able to demonstrate that this is technically feasible and that it is cost effective since the overall cost of the decontamination-recycling-reuse route is still lower than the disposal and replacement route. Moreover, it saves natural resources and decreases the radioactive waste volumes.

This choice implies the setup of a strong Quality Assurance program to guarantee the traceability and pushes the industry to develop cost effective decontamination and measurements techniques.

A major effort must still be done to harmonize the different regulations and to fix "reasonable" clearance levels.

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Nuclear and non-nuclear safety aspects in Nuclear Facilities dismantling,
The example of a PWR pilot decommissioning project

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Abstract. The dismantling of nuclear facilities, and in particular of nuclear power plants, involves new challenges for the nuclear industry. Although the dismantling of various activated and contaminated components is nowadays considered as almost industrial practice, the safety aspects of decommissioning bring some specific features which are not always taken into account in the operation of the plants. Moreover, most of the plants and facilities currently decommissioned are rather old and were never foreseen to be decommissioned. The operations involved in dismantling and decontamination, often imply new or unforeseen situations.

On the nuclear, or radiological side, the radioprotection optimisation of the operations involved often requires to model the environment and to analyse different scenarios to tackle the operation. Recent 3-D software (like the Visiplan software) allowing to represent the actual environment and the influence of the various sources present, is really needed to be able to minimise the radiological impact on the operators. The risk of contamination spread, by opening loops and components or by the dismantling process itself, is also an important aspect of the radiological protection study.

Nevertheless, the radiological aspects of the safety approach are not the only ones to deal with when decommissioning nuclear facilities. Indeed, classical industrial safety aspects are also important: the dismantling can bring handling and transporting risk (heavy loads, difficult ways, uneasy access, etc.) but also the handling of toxic or hazardous materials. For instance, the removal of asbestos in contaminated areas can lead to additional hazard; the presence of alkali metals (like Na or NaK), of toxic metals (like e.g. Beryllium) or of corrosive fluids (acid,...) have to be tackled often in unstructured environment, and sometimes with limited knowledge of the actual situation. This leads to approach the operations following the ASARA principle (As Safe As Reasonably Achievable) instead of the rather restricted ALARA principle.

1. Introduction

The BR3 (for Belgian Reactor Nr 3) was the first PWR installed in Europe. In service since 1962, it was shutdown in 1987 after 25 years of operation. It is a quite small reactor with a thermal power of 40.5 MW_{th}. At the end of its operating life, the European Commission, in the framework of its five-year plan of RTD, selected in 1989 the BR3 as one of the four pilot decommissioning projects on decommissioning of nuclear installations.

The main steps of the decommissioning programme up to now were:

- Full System Decontamination of the primary loop in 1991,
- Dismantling of the high active thermal shield with three different techniques, of the Vulcain internals and of the first set of Westinghouse internals (30 years decay time) by mechanical cutting up to 1996,
- The dismantling of the auxiliary circuits started in 1995 and is still going on,
- The dismantling of the reactor pressure vessel ended in 2000,
- The construction and start of the exploitation of thorough decontamination processes for dismantled pieces in the period 1996 to 1999.

For the dismantling of the contaminated loops, we decided to reduce the dose rate of the loops by chemical decontamination, to dismantle the pieces and to sort them following their specific evacuation route and, finally, to thoroughly decontaminate the pieces with the objective to minimise the amount of materials disposed of as radioactive waste.

Moreover, the clean up of the whole research centre, financed by a technical liabilities funding set up by the Belgian Government in 1989, involved also the dismantling and decommissioning of hot cells and contaminated research laboratories, some of them having even led to the official release for unrestricted reuse of the building.

More than 10 years of practical experience on decommissioning has thus been accumulated through these projects. The BR3 decommissioning project was not only selected as European pilot project but was also considered as Belgian pilot for the Authorities and as SCK•CEN pilot project for the complete implementation of the ALARA principle and for an integrated approach of industrial and radiological safety.

2. The specific safety aspects and the new challenges of decommissioning

The specific aspects of safety in decommissioning will be regarded at two different levels: the radiological one and the industrial one. Then we will analyse the potential influence of one aspect on the other. Decommissioning involves indeed new challenges as it requires working in a continually changing environment, on operations for which the installations were not foreseen, and often with poor or scarce documentation on the facilities to be handled.

2.1. The radiological aspects

The main problems or issues faced when dealing with decommissioning are probably the following ones:

- required presence in areas with potentially high radiation field
- opening of loops and piping with internal contamination
- continuously changing environment by the dismantlement of parts of the installation, sometimes including shielding
- potential unforeseen situation due to the quality of the remaining documentation and to the fact that the installations were often never foreseen to be decommissioned
- "one-shot" operations, implying sometimes heavy preparation to be distributed only to one operation, thus economically less interesting than repetitive operations.

All these situations were encountered in the BR3 pilot decommissioning project, and led to the implementation of a formal ALARA procedure to deal with such kind of issues.

But the optimization of the radioprotection should take complete operations into consideration, in order to avoid transferring the radiological risk from one operation to the next one or to a future operation in the follow up of the project. Let us illustrate this by two examples.

The first one concerns the decontamination of the primary loop and auxiliary circuits. This operation concerned the whole primary loop, including the reactor pressure vessel. The operation itself implied a significant dose uptake for the operators, mostly because the reactor head had to be closed, and this activity is still performed "by hands" on this old reactor generation. The total dose uptake for the operation was about 160 man-mSv (preparation and post decontamination operations included), but allowed to save between 4 to 7 man-Sv for the future dismantling of the primary loop (carried out almost 10 years later). The dose rate in the vicinity of the loop was reduced by a factor of at least 10, leading to an ambient dose rate between 60 and 80 μSv/h.

This shows that the optimisation of the radioprotection must take future operations into account for getting the whole scene.

The second one concerns the optimisation of the dismantling of the primary loop piping and all auxiliary equipments situated in the primary loop area. This operation was quite complex, involved a lot of equipments and components, and let various potential alternatives open for the chronology and working procedure.

The SCK•CEN developed at the end of the 90's a user-friendly optimisation tool (so called "Visiplan") which allows to analyse rapidly, and in a 3D graphical way, different scenarios and procedures. The area was then simulated (see fig.1.), and different alternatives were envisaged. The use of this efficient tool allowed to optimise the procedure and to take into account the changing environment of the plant (fig.2 and 3).

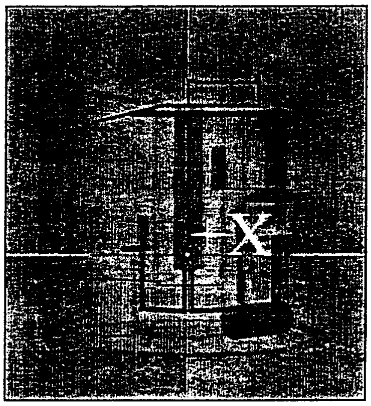

Fig.1: Schematic representation of the operation room

Fig 2 and 3: Primary loop area before and after dismantling

2.2. *The industrial safety aspects*

It is obvious that a decommissioning operation implies several aspects of classic industrial safety. Such an operation can be compared to building construction yards, with additional constraints coming from the radiation and contamination protection aspects (like e.g. the wearing of gloves, overshoes, overall, full-face mask, pressurised suit, shielding, etc..). The classic risks of falling, load handling accidents, cutting tools wound etc. are obviously present in a decommissioning yard. Nevertheless, the decommissioning of a nuclear facility presents also some specific safety aspects which must also be taken into account.

The changing environment is the first one, as protective equipments can be dismantled, infrastructure equipments can be removed and sometimes structural integrity must also be regarded. Moreover, the operators working on a dismantling yard are not always fully aware of the new risks and dangers which can be encountered; this is mostly the case when former nuclear operators of the dismantled plant are mobilized for this new type of activity.

A second important aspect is the potential presence of toxic or dangerous materials present in the dismantled plant; like e.g. asbestos in thermal insulation or in cement, acids or chemicals remaining in tanks and piping or used as decontaminating agent. It is often needed to train the operators for being able to face these risks and to know which protection has to be taken.

The combination of industrial and radiological risks can also be a specific aspect of the operation, leading to adapted procedures and methods and to look for the way of minimum risk instead of focusing only on the radiological aspect of the safety. This will be developed in the next paragraph.

2.3. The influence of one safety aspect on the other one, and the ASARA approach

Sometimes, the influence of one safety aspect on the other can be important, and one should then apply a broader concept than the ALARA principle to improve the safety of the operators and the environment. This will be highlighted by two typical examples from the BR3 dismantling project.

The first one is a quite typical case, where the radiological aspect was only considered by the operator. It was given the name of the "ladder syndrome" by our internal health physics department. Some operator who had to work in a controlled area with a ladder was wondering about the potential contamination of his tool. His ladder was made out of aluminium, so the decontamination should easily be performed afterwards. Then he considered that the rubber pieces placed at the extremities of his ladder should give rise to problems for further decontamination. These safety-related rubber pieces were placed to avoid the ladder to glide during use. Our worker decided to put some little plastic bags around the rubber extremities to protect them against the potential contamination. Then he went into the controlled area and used his tool, that glided! The worker fell and one of his legs was broken. He had to stay home for three months! This is a very simple example of interaction between radiological and non-radiological hazards. There are many other cases. The question is then "By optimisation of the radiological side of our works, do we not transfer some risks to the non radiological field?"

The second one concerns the removal of asbestos in places difficult to access. Elimination of the thermal insulation from the legs of the reactor pressure vessel at BR3. This set of operations took place during the month February 1999. Workers belonging to the staff of BR3 performed the related tasks. This was the result of preliminary discussions with the administration of the Technical Inspection on the Workplaces. Indeed, due to the limited quantities of asbestos which had to be taken away and to the very well defined tasks to be performed, she concludes that the nuclear know-how and the safety features usually applied were adequate and that the managers of the BR3 didn't have to work with a licensed external firm. The estimated collective dose was 3,3 man-mSv and the received collective dose was 2,8 man-mSv (9 days and 10 workers).

Another asbestos removal yard was performed for the insulation of the primary loop. The work was performed as foreseen by a specialised and licensed company but some modifications were brought and some requirements were added too. For example, instead of working "top-down" as far as the removal of asbestos was concerned, the contractor accepted to begin in the middle of the steam generator. This decision was taken as a result of the pre-job ALARA study performed with the software VISIPLAN. Another example is related to the use of the full-face masks. After a few days, due to our additional check-up for internal contamination, it seemed that the masks in use in nuclear areas were more efficient and that they were more guaranteed for the safety of the workers. A third example copes with "at random" check-up in the Whole Body Counter for potential internal contamination. This has led to the evidence of malfunction (or misuses) of the personal protection equipment and brought to more severe requirements. On the other side, this supplementary measurement was very well accepted by the workers. Their fear, as far as nuclear risks were concerned, has really decreased! Finally, the dose for each worker was daily recorded and transmitted to the partners (contractor, BR3, Health Physics and Safety Department). Finally the operation took 35 days (instead of 50 as estimated) and the cumulative dose uptake amounted to 19.2 man-mSv instead of 89 as estimated.

These aspects of mutual influence of industrial and radiological safety, as well as new trends indicate the need for an extension of the ALARA approach to a broader extend, i.e. introducing the concept of ASARA principle, for As Safe As Reasonably Achievable.

3. Conclusions

The dismantling of nuclear installations is leading to several new challenges on technical side but also on the safety approach. The mutual influence of radiological and industrial safety on decommissioning yards have led to introduce the ASARA approach, for reducing and minimizing the total risk for the operators and the environment. These aspects were applied practically at the BR3-PWR pilot dismantling project.

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STUDIECENTRUM VOOR KERNENERGIE
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Belgoprocess

ONDRAF

Press Notice

Brussels/Dessel/Mol, 12 September 2002.

The dismantling of the BR3 reactor enters a new phase: the transfer of the spent fuel to a dry storage place on the Belgoprocess site.

The decommissioning of the BR3 reactor, one of the two main current decommissioning projects in Belgium enters a new phase: the evacuation of the spent fuel still present in the plant. This operation is necessary to continue the decommissioning operations. The spent fuel is loaded in specially shielded casks and is then transferred from the SCK•CEN site to the Belgoprocess site, situated at about 500 m distance. The transport is carried out by Transnubel, a company specialised in radioactive material transport, which has the necessary licenses. On the site of Belgoprocess, the casks are safely stored in a new building (bldg. 156X) awaiting a definitive solution for their long term management. The first transport took place on 16 September, and will be followed by 6 other transports.

Reminder: the BR3, first European PWR reactor, started in 1962, was definitely shut down in 1987. In 1989, this plant has been selected by the European Commission as European pilot decommissioning project. The dismantling operations began by the decontamination of the primary loop. The BR3 team has then cut under water the highly radioactive reactor internals to allow their removal and conditioning. In 2000, the BR3 team cut, also under water, the reactor pressure vessel, and started in parallel the dismantling of various loops of the plant. Afterwards, large equipments, like the steam generator and the pressurizer will be decontaminated and dismantled. The whole decommissioning operation should be completed in 2009, at which date the reactor and the building would be completely dismantled.

In order to continue the dismantling operations, it is absolutely necessary to remove the spent fuel elements, still present in the plant. Therefore, the fuel assemblies are placed in specially designed and shielded casks (type CASTOR-BR3, from the German company GNB) weighing each 28 t. From mid September, these casks will be transported by road from the SCK•CEN site to the Belgoprocess site, with a special trailer (on a distance of about 500 m). On the Belgoprocess site, the casks are stored in a new building (bldg 156X) where the control and monitoring will be continuously assured awaiting an adequate definitive solution, like e.g. deep geological disposal. In total, seven transports – one cask per transport – are foreseen, until November 2002. The casks, the storage building, the loading and the transport have been documented, analysed and approved (licensed) by the Belgian authorities.

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Dismantling of the BR3 Reactor Pressure Vessel

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052/02-21

EPRI International Decommissioning and Radioactive Waste Workshop
Dounreay Site – Thurso, Scotland
September 17-19, 2002

ABSTRACT

The BR3 PWR is a small nuclear power plant (thermal power 40.9 MW_{th}, net electrical power output 10.5 MWe), designed in the late fifties and started in 1962. It was definitely shut down in 1987.

In 1989 the BR3 was selected by the European Union as pilot decommissioning project in the framework of its RTD programme on the decommissioning of nuclear installations. A pre-dismantling decontamination of the reactor primary loop was carried out and allowed to save doses to the operators. The savings are estimated to be up to about 4 to 7 man-Sv.

The decommissioning project concerns mainly:

- The dismantling of the highly radioactive reactor internals. Different techniques were used and compared on a first actual piece called the thermal shield: from plasma arc torch cutting to mechanical sawing, including also electric discharge machining. Based on the experience gained during this part of the project, the mechanical cutting techniques were promoted for the segmentation of both sets of internals, the separation and the segmentation of the RPV.
- For the dismantling of the reactor pressure vessel, wet and dry dismantling were studied and compared. For economical and feasibility reasons, the wet dismantling was selected. Afterwards, two underwater segmentations were also studied: in-situ segmentation and a segmentation after having removed the RPV out of its cavity.
- Mainly for technical reasons, the reactor pressure vessel was removed in one piece out of its cavity in order to be cut in the former refuelling pool. The disconnection of the RPV from the other parts of the plant was followed by the reinstallation of the water tightness of the pool in order to allow remote underwater segmentation. The disconnection, the water tightness reinstallation and the segmentation represented important challenges. The subtasks will be extensively described in the paper: disconnection from the pools floor, removal of the thermal insulation from the legs, decoupling from the primary loop at two levels, from its supporting structure, the reinstallation of the water tightness of the pool and testing, the removal of the RPV out of its cavity, the remote dismantling of its surrounding thermal insulation (which led to an annoying pool water turbidity) and, finally the effective RPV dismantling.
- For the segmentation, two main cutting equipments were used: the milling cutter for cutting the RPV into rings and the band saw machine for cutting each ring into segments. The band saw machine was also used in order to cut the RPV upper flange into pieces vertically as well as horizontally.
- The last generated pieces, the highest radioactive ones, were evacuated at the end of 2000.
- Waste characterisation, minimization and management is an important part of the task in order to reduce evacuation and storage costs.
- ALARA approach was applied from the early beginning of the project.
- For each "key operation" cold tests were organized in order to optimize the work and to take benefit of the learning effect of such operation.

Results of the operations will be presented, the lessons drawn for the technical choices, dose uptake minimization, waste reduction and the technical problems met will be highlighted.

As a pioneering project, the dismantling of the BR3 Reactor Pressure Vessel has shown the technical feasibility of such an operation in a safe and economical way as well.

INTRODUCTION: THE BR3 DECOMMISSIONING SUMMARY

BR3 is a small 10 MWe PWR shutdown in 1987 after 25 years of operation. It was selected as one of the four pilot projects of the EU for its R&D programme on Decommissioning of nuclear installations. The decommissioning project started in 1989. In 1991, a Full System Decontamination of the primary loop reduced the dose rate in the vicinity of the primary loop by a factor 10. The same year, a first high active internal, the 5.4 t thermal shield was dismantled underwater by 3 different dismantling techniques: the EDM cutting, the milling cutter and the plasma arc torch. Mechanical cutting, essentially milling cutter and band saw, were selected for the further dismantling of the two sets of internals; the original Westinghouse internals ("33 years decay") and the Vulcain internals ("7 years decay"). This allowed comparing deferred dismantling with immediate dismantling. No significant radiological, technical or economical profit was gained by dismantling the old internals because due to the still high dose rate of 2 to 3 Sv/h at mid plane, remote underwater cutting was still required.
The next important step was the cutting of the 28 t Reactor Pressure Vessel. The produced waste has been sent to the Belgian waste conditioner for conditioning and interim storage. Only three pieces of the vessel still remain: the bottom, the head and the insulation shroud. They will be cut in the next phase with a new technique, the high-pressure water jet cutting technique.
Besides the dismantling of high-activated pieces, the dismantling of some contaminated circuits was also performed using mostly hands on cutting techniques. Minimizing the amount of radioactive waste and free release of the dismantled materials have always been our main objectives. Recycling of slightly radioactive metallic materials could be performed thanks to an agreement with a nuclear foundry. For concrete, an R&D programme has been started to recycle radioactive concrete in the radioactive waste conditioning sector. Progress was also made on the establishment of free release procedures and on the development of decontamination techniques for metals and concrete [1] [2].

THE DISMANTLING OF THE VESSEL

This paragraph discusses first the selection of the strategy. Two main strategies, namely dry and wet cutting, were compared. The paragraph describes further the different steps of the vessel dismantling. The steps to be discussed start from the vessel's separation from the primary circuit through the cutting of the vessel and removal of the waste materials.

Strategy selection

A detailed study for the complete dismantling of the RPV in air or under water has been carried out. Based on the results of the preceding projects, the mechanical cutting processes were first promoted and analysed.
For the comparison of dry and wet cutting, the study focused on the following areas:

- the technical feasibility;
- the radiation protection and safety of the operators, including the case of equipment failure;
- the shielding needs for coping with the radioprotection requirements.

The studies allowed to foresee globally the manpower, operation duration and costs of both operations. For the dismantling of the RPV, the underwater method was finally selected. The RPV being surrounded by an annular Neutron Shield Tank (NST), the vessel can be submerged and only the three penetrations for the primary loop pipings have to be closed to assure the leak tightness of the pool during the operation.

Further design for the RPV dismantling led to analyse two different approaches: the in-situ dismantling, where the RPV remains in place (under the bottom of the refuelling pool) while being cut into rings, and the "one-piece removal", where the vessel is removed in one piece into the refuelling pool, and then segmented into pieces ready for packaging.

The advantage of the latter is the accessibility of the RPV and its insulation shroud from the outside, giving the possibility to reuse the dismantling tools and equipments designed for the internals dismantling. Moreover, this approach simplifies greatly the dismantling of the RPV insulation shroud situated at about 100 mm outside the vessel wall.

Preliminary operations
These operations (**see figure** 1) were executed with a dry refuelling pool, the RPV still being in its cavity under the bottom of the refuelling pool. So the access to the pool floor was possible but had to be minimised as much as possible for radioprotection reasons.

Disconnection of the RPV from the bottom of the reactor pool
The selected process for cutting the bottom of the reactor pool is the plasma arc torch handled by an operator. The cutting has to be done quickly for limiting the dose uptake of the operations (radioprotection optimization). In addition to this separation, different cuts at the bottom of the reactor pool were also needed to give access to the fastening bolts of the RPV support flange, to give access to the hot and cold legs thermal insulation and to allow the installation of the sealing equipment for the future water tightness of the pool.

Removal of the asbestos situated around the primary pipes near the RPV
This operation was carried out by SCK•CEN personnel, as the nuclear hazard was estimated to be far above the asbestos hazard. Nevertheless, to avoid the spread of asbestos fibers, a double confinement was installed in the RPV pool.

Disconnection of the RPV from the hot and the cold legs
- *Cutting of the primary pipes at the outside of the bioshield.* The main operation is the cutting of the pipes at the RPV flange level. But regarding the very tight space available to perform this operation, access was needed through the primary pipes at the bioshield side. This operation was carried out with a quite common automatic pipe cutter, using two lathe tools diametrically opposed.
- *Cutting the primary pipes near the RPV.* This operation was delicate due to the fact that access was only available at the inside of the piping. We thus developed, with an industrial partner, an automatic milling cutter able to cut the necessary thickness. The challenge was to have a machine fitting into a diameter of 254 mm, able to cut up to 110 mm wall thickness. Finally, it was decided to make a second cut of the primary pipe connections just above the support flange of the RPV in order to get access to all the RPV fastening bolts. The cutting tool is an automatic milling cutter with a diameter of 30 mm for the first part of the cut, 25 mm for the second, deepest, part.

Separation of the RPV from the NST
The selected procedure to remove the 24 fastening bolts of the RPV on the Neutron Shield Tank is the pneumatic unbolting. Due to the high level of corrosion, this operation took about three times more than foreseen.

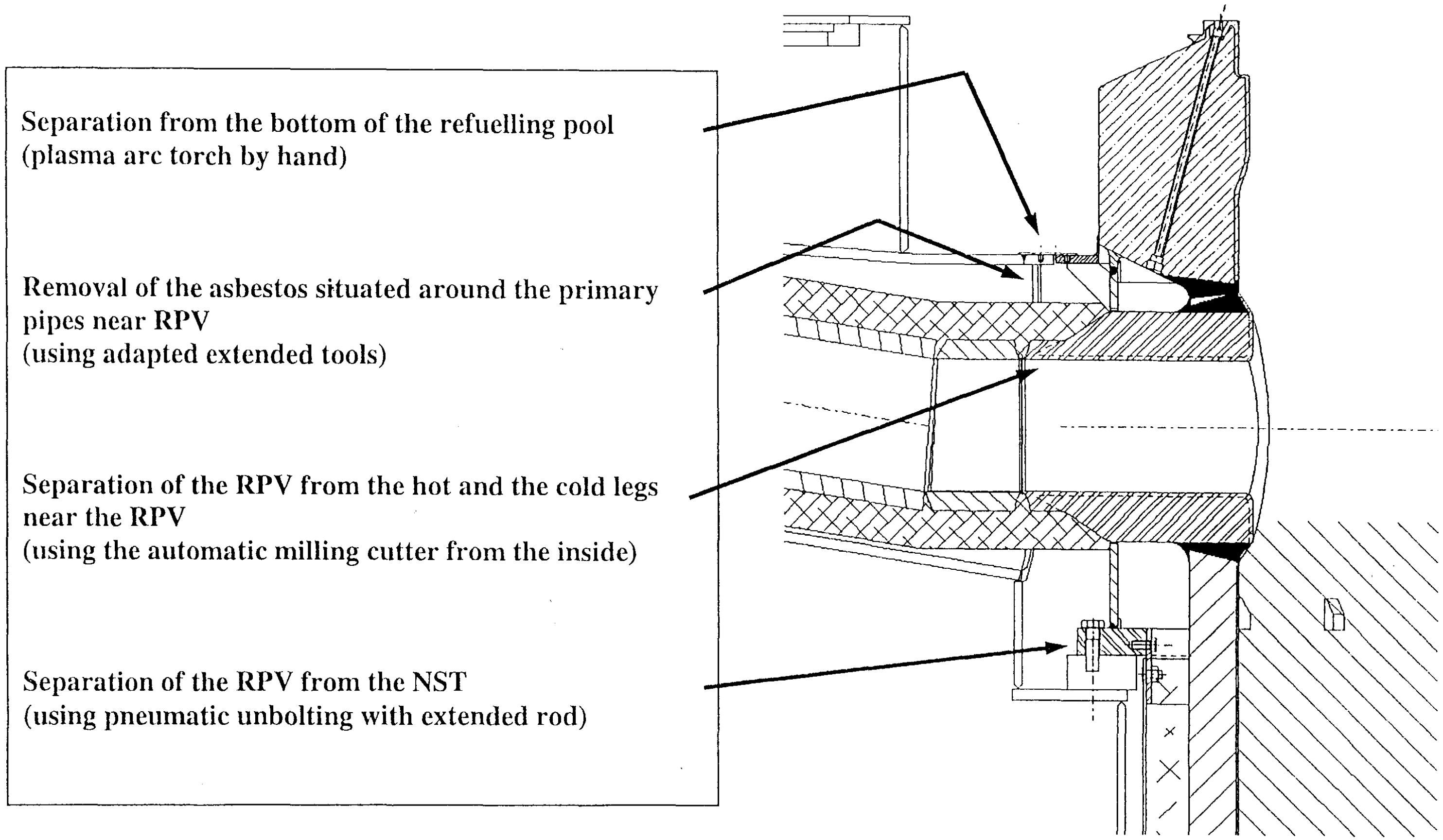
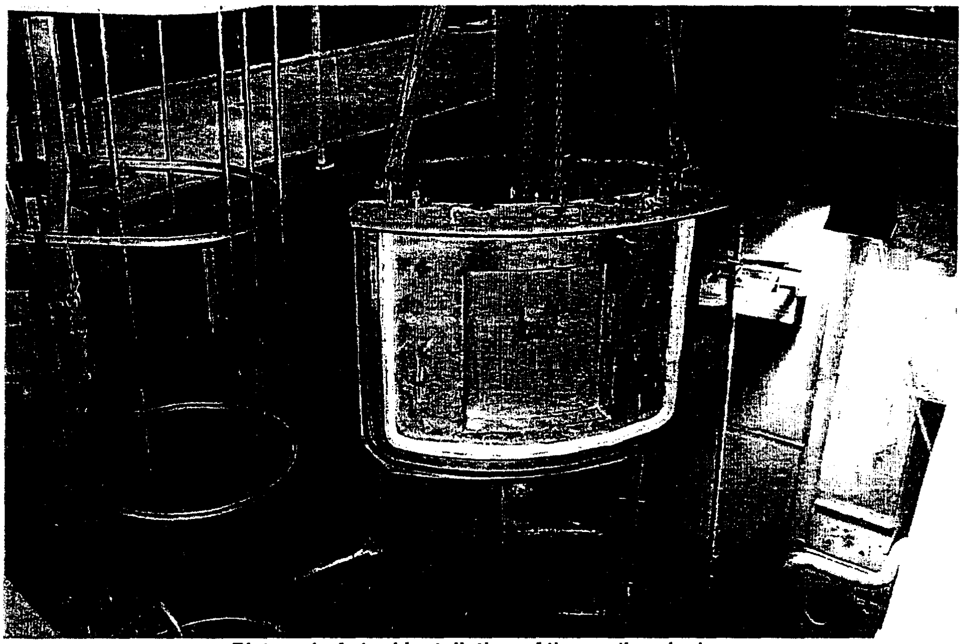


Figure 1: The separation steps of the reactor pressure vessel

Reinstallation of the water tightness of the NST and the reactor pool
As the RPV and its primary pipes were part of the pool leak tightness system, we had to seal the openings made by the primary pipe cutting, which is a very tight space. The operation was carried out with an industrial partner, who developed a system based on an epoxy-based polymer and form-shaped sealing system. Cold testing was carried out on real scale mock up and everything was ready for the installation (**Picture 1**). During the installation, a major positioning problem raised. More about this problem further on in the paper.



Picture 1: Actual installation of the sealing device.

Finally the RPV was ready to be lifted. A guiding system was also installed as the mechanical clearance between the RPV and the sealing devices was less than 10 mm. On August 24, 1999 the pressure vessel (28 ton) was then lifted up in one day, using a new gantry crane installed above the RPV pool. The water level in the pool was raised at the same pace as the RPV lifting.

Removal of the insulation shell
The insulation shell is bolted to the RPV through two profiles and on the upper side it is bolted to the RPV supporting skirt. It was necessary to remove 60 bolts to free the insulation shell from the RPV. Because of the horizontal position of these bolts they would be drilled by a remote hydraulic hole cutter. For reaching easily the different levels at which the bolts were placed, the remote hydraulic hole cutter can move up and down along a beam. Here again, cold tests were organized.
During the execution of this dismantling task, we encountered two problems. First of all, there was a positioning problem for the cutting tool and second, there was a visibility problem with the pool water. A detailed explanation of these problems follows in a next chapter.

Removal of the insulation and the fastening profiles of the insulation shell
The insulation shell was bolted on the RPV by T-shaped fastening profiles and connection pieces on two levels. Between and on top of these fastening profiles there is a thermal glass fiber insulation, fastened with a metal mesh. The insulation was also held together with metal strips. On the bottom side of the RPV the insulation is tightened to the RPV with eight strips. The strips are attached on the RPV by bolts throughout the insulation material.
As the mesh was totally rusty, the removal of the insulation was done using a long handling tool. The liberated insulation fell into a fishing net previously installed on the floor of pool. By remotely closing the fishing net, the insulation was taken out of the water and removed as standard low level waste.

The T-shaped fastening profiles of the insulation shell were the last items that had to be removed before the actual cutting up of the vessel could take a start. First, it was foreseen to unscrew the bolts of these fastening profiles. But due to the strong corrosion of the bolts, these profiles could easily be ripped of the vessel with a hook. As the fastening profiles are low activated, their further dismantling was done by hand.

Dismantling operations of the RPV

Cutting of the RPV into rings

After the completion of the study, the adaptation pieces, the positioning and the clamping devices were ordered and fabricated. **Figure 2** shows the different levels of the horizontal cut.

These cuts will be executed by a circular saw available at BR3. The tests in the pool on a mock-up of the RPV scale (scale 1/1) were programmed for validation of the cutting parameters. Cut 1, of the bottom of the RPV, was the most difficult one and the first clamping system was not perfect (lot of vibrations during cutting). This problem led us to a design review. Some additional clamping devices were added and the cutting procedure was adapted.

For the RPV flange cutting, a band saw system was used.

Cutting the rings into segments

These cuts were done by a band saw used before for the dismantling of the reactor internals.

The tests in the pool on a mock-up of the RPV (scale 1/1) were also programmed for validation of the cutting parameters. The most difficult cut was the one through the RPV flange. Nevertheless, on the mock-up, we succeeded to cut at the first time.

ENCOUNTERED PROBLEMS AND SOLUTIONS

During the whole dismantling phase of the RPV, the team encountered two major problems. These problems concerned on one hand some "non conformities" or discrepancies with the "as built" drawings and, on the other hand, severe turbidity problems of the cutting pool.

- *Problems with the "non conformities" of as built drawings.*

As already explained earlier in the text, one had to reinstall the water tightness of the reactor pool. This would be done with three special designed sealing devices. Already in the very beginning, one had to stop the operation because it was impossible to position the sealing devices due some discrepancies between the "as built" drawings and the reality. This problem is a common problem in the dismantling of old nuclear facilities. Therefore, the design of the sealing devices had to be revised, and the sealing devices themselves had to be adapted. The positioning of these devices was finally carried out in June 1999 instead of March 1999.

Another problem of the same category is the one encountered with the insulation shell removal. Starting the work, it became almost impossible to locate the screw heads due to a high level of corrosion on the shroud surface. Therefore it was impossible to locate these bolts for cutting them with a hole cutter machine. It was then decided to cut the entire circumference of the core shroud at the corresponding level of the bolts. This work method required 10 times more holes to be cut than foreseen.

- *The turbidity problem of the pool*

When we removed the insulation shell (a protective metal sheet for the thermal insulation situated around the RPV) a major problem occurred: significant water turbidity appeared. This was due to the thermal insulation which became breakable into something looking like dust but also to rust. Sometimes, the visibility was so bad that the operation had to be stopped. Additional filtration and purification facilities were installed to solve this problem. Resolving this visibility problem led to an extra burnable waste volume of 0.4 m^3 of filters.

The same problem appeared again when we carried out the horizontal cut with the band saw for the removal of the vessel flange. Remaining insulation, situated under the vessel flange came again in the pool water and caused a new turbidity problem. Nevertheless, due to the presence of the additional filtering equipment, the water visibility could be recovered in a rather fast way.

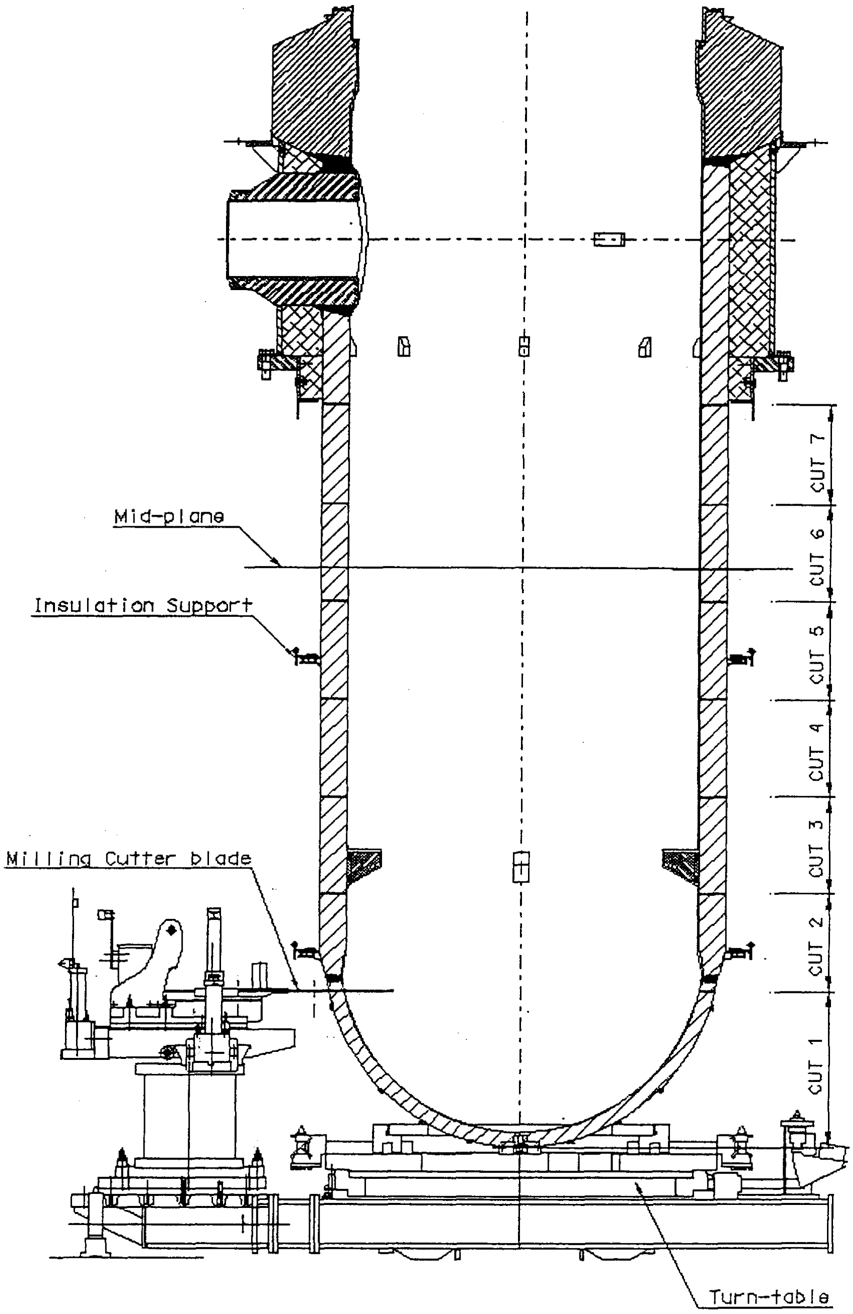


Figure 2: The BR3 circular saw will carry out the horizontal cuts

WASTE

It is the Belgian National RadWaste Authority (ONDRAF/NIRAS) who sets up the different acceptance criteria on waste types and waste packages. Concerning the solid waste (big pieces), there are three important groups of waste and the distinction between these groups is based on the contact dose rate. These are Low Level solid waste (LLW) with a contact dose rate < 2 mSv/h, Medium Level solid waste (MLW) with a contact dose rate between 2 mSv/h and 0.2 Sv/h and High Level solid waste (HLW) with a contact dose rate > 0.2 Sv/h.
Concerning the waste packages, there are only two different types, namely the standard 400l drum and the standard 200l drum. The 400l drum is used for the big pieces. At the waste facility these drums will be filled with grout. The 200l drum is used for small pieces and is foreseen for the supercompaction.

The vessel itself led to the production of a high volume of waste and more particularly high and medium activated waste .
For radiation protection reasons, this waste had to be manipulated under water. Therefore we used a system similar to the one used during the dismantling of the reactor internals; but, based on the gained experience, some modifications were carried out. The waste removal system existed out of two racks, an upper and lower rack. Both racks could be bolted together to fit in a 400l drum. The replacement of the carrier structure by 4 bolts was the main modification that was carried out, leading to the following advantages:

- weight saving of the 'death' mass,
- only one lifting device instead of two with the previous system, and
- an easier underwater manipulation of the racks (saves time and dose).

In total, nine transports were carried out to the Belgian waste conditioner and intermediate storage facility representing a volume of 3.6 m³ on high-activated waste (including high-activated swarfs). These transports were done in the second half of 2000 at a transportation rate of one transport every two days.
The medium activated waste was manipulated with the same rack system and represents a volume of 4.8 m³.
Low activated waste, mainly the vessel flange and the bottom ring led to a volume of 6.8 m³. The low activated secondary waste like swarfs, cutting blades and filters were spread out over several waste drums.

CONCLUSIONS

Being a pilot project, BR3 had demonstrated the feasibility of the dismantling of high-activated pieces using its two sets of internals. BR3 learned a lot about remote cutting techniques, remote operations and waste handling.
Using this experience, the BR3 team extrapolated his knowledge in order to demonstrate the feasibility of the same operation on the RPV of a PWR plant. As the used cutting techniques are now well known, the biggest part of the challenge was concentrated on the preparation works: to separate the RPV from the rest of the facility and to bring it under water into the refuelling pool which was used as underwater cutting workshop.

As these works are finished, we can conclude that the cutting of the primary loop from the inside and the reinstallation of the water tightness of the pool were important challenges.
BR3 completed the cutting phase of its Reactor Pressure Vessel. This operation was terminated just before the summer holidays of 2000, respecting the overall project planning. From a technical point of view, no significant problems were encountered which means that the BR3-team mastered the mechanical cutting techniques for the dismantling of high radioactive structures.
With the final dismantling study of the NST, a new challenge starts as a quite new cutting technique, the high pressure water jet cutting, will be used combined with a telemanipulated arm.

Even if BR3 is a PWR type plant, the SCK•CEN experts still think that the results and the lessons learned can be used to derive data for all kinds of power plants dismantling activities.

References

[1] RPV and Internals Dismantling Project (BR3, EWN, KRB-A), Research Contract FI4D-CT95-0001, Progress Report January-June 96, Ref. 59/96-55

[2] RPV and Internals Dismantling Project (BR3, EWN, KRB-A), Research Contract FI4D-CT95-0001, Progress Report January-June 97, Ref. 59/97-30

[3] RPV and Internals Dismantling Project (BR3, EWN, KRB-A), Research Contract FI4D-CT95-0001, Progress Report January-June 98, Ref. 212/98-15

[4] RPV and Internals Dismantling Project (BR3, EWN, KRB-A), Research Contract FI4D-CT95-0001, Progress Report July-December 1998, Ref. 212/99-06

[5] "1999 Summer School on Radwaste and Decommissioning", Cambridge, June 1999, V. Massaut

附件五

比利時核設施拆除混凝土
Free Release 標準

Official clearance levels in Belgium.

b) Lorsqu'il s'agit de mélanges de radionucléides, il y a lieu de veiller au respect de la condition suivante :

$\Sigma_j C_j / C_{j,L} \leq 1$

où :

C_j est l'activité spécifique du radionucléide j dans le déchet et $C_{j,L}$ est le niveau de libération figurant dans le tableau A pour le radionucléide j.

En cas de mélanges parents – descendants, les produits de filiation peuvent être négligés dans cette formule s'ils ont été pris en compte lors de la fixation des niveaux et n'excèdent pas l'activité à l'équilibre.

Tableau A - Niveaux de libération

Nucléide [1]	Niveaux de libération [2, 3] [kBq/kg]
H-3	100
Be-7	10
C-14	10
Na-22	0.1
P-32	100
P-33	100
S-35	100
Cl-36	1
K-40	1
Ca-45	100
Ca-47	1
Sc-46	0.1
Sc-47	10
Sc-48	0.1
V-48	0.1
Cr-51	10
Mn-52	0.1
Mn-53	1000
Mn-54	0.1
Fe-55	100
Fe-59	0.1
Co-56	0.1
Co-57	1
Co-58	0.1
Co-60	0.1
Ni-59	100
Ni-63	100
Zn-65	1
Ge-71	10000
As-73	100
As-74	1
As-76	1
As-77	100
Se-75	1
Br-82	0.1
Rb-86	10
Sr-85	1
Sr-89	10
Sr-90+	1
Y-90	100
Y-91	10
Zr-93	10
Zr-95+	0.1
Nb-93m	100
Nb-94	0.1
Nb-95	1
Mo-93	10
Mo-99+	1
Tc-96	0.1
Tc-97	10
Tc-97m	10

Nucléide [1]	Niveaux de libération [2, 3] [kBq/kg]
Tc-99	1
Ru-97	1
Ru-103+	1
Ru-106+	1
Rh-105	10
Pd-103+	1000
Ag-105	1
Ag-108m+	0.1
Ag-110m+	0.1
Ag-111	10
Cd-109+	10
Cd-115+	1
Cd-115m+	10
In-111	1
In-114m+	1
Sn-113+	1
Sn-125	1
Sb-122	1
Sb-124	0.1
Sb-125+	1
Te-123m	1
Te-125m	100
Te-127m+	10
Te-129m+	10
Te-131m+	0.1
Te-132+	0.1
Te-134	0.1
I-125	1
I-126	1
I-129	0.1
I-131	1
Cs-129	1
Cs-131	1000
Cs-132	1
Cs-134	0.1
Cs-135	10
Cs-136	0.1
Cs-137+	1
Ba-131	1
Ba-140	0.1
La-140	0.1
Ce-139	1
Ce-141	10
Ce-143	1
Ce-144+	10
Pr-143	100
Nd-147	10
Pm-147	100
Pm-149	100
Sm-151	100
Sm-153	10
Eu-152	0.1
Eu-154	0.1
Eu-155	10
Gd-153	10
Tb-160	0.1
Dy-166	10
Ho-166	10
Er-169	100

Nucléide [1]	Niveaux de libération [2, 3] [kBq/kg]
Tm-170	10
Tm-171	100
Yb-175	10
Lu-177	10
Hf-181	1
Ta-182	0.1
W-181	10
W-185	100
Re-186	10
Os-185	1
Os-191	10
Os-193	10
Ir-190	0.1
Ir-192	0.1
Pt-191	1
Pt-193m	100
Au-198	1
Au-199	10
Hg-197	10
Hg-203	1
Tl-200	1
Tl-201	10
Tl-202	1
Tl-204	10
Pb-203	1
Pb-210+	0.01
Bi-206	0.1
Bi-207	0.1
Bi-210	10
Po-210	0.01
Ra-223+	1
Ra-224+	1
Ra-225	1
Ra-226+	0.01
Ra-228+	0.01
Ac-227+	0.01
Th-227	1
Th-228+	0.1
Th-229+	0.1
Th-230	0.1
Th-231	100
Th-232+	0.01
Th-234+	10
Pa-230	1
Pa-231	0.01
Pa-233	1
U-230+	1
U-231	10
U-232+	0.1
U-233	1
U-234	1
U-235+	1
U-236	1
U-237	10
U-238+	1
Np-237+	0.1
Np-239	1
Pu-236	0.1
Pu-237	10

Nucléide [1]	Niveaux de libération [2, 3] [kBq/kg]
Pu-238	0.1
Pu-239	0.1
Pu-240	0.1
Pu-241	1
Pu-242	0.1
Pu-244+	0.1
Am-241	0.1
Am-242m+	0.1
Am-243+	0.1
Cm-242	1
Cm-243	0.1
Cm-244	0.1
Cm-245	0.1
Cm-246	0.1
Cm-247+	0.1
Cm-248	0.1
Bk-249	10
Cf-246	10
Cf-248	1
Cf-249	0.1
Cf-250	0.1
Cf-251	0.1
Cf-252	0.1
Cf-253	1
Cf-254	0.1
Es-253	1
Es-254+	0.1
Es-254m+	1

Notes

(1) Pour les radionucléides indiqués par le signe +, les descendants figurant au tableau B de la présente annexe IB ont été pris en compte dans les calculs des niveaux de libération.

(2) Les niveaux se rapportant aux radionucléides naturels, représentés ici sur fond gris, ne sont pas d'application pour les activités professionnelles autorisées en application de l'article 9, sauf décision contraire de l'Agence.

(3) Le respect de ces niveaux n'est pas suffisant dans le cas des établissements utilisant des substances radioactives de période inférieure à 6 mois; dans ces cas, il y a lieu de se référer aux dispositions de l'article 35.

Tableau B - Descendants pris en compte dans les calculs de niveaux de libération pour les radionucléides indiqués par le signe + au tableau A

Parent	Descendance
Sr-90	Y-90
Zr-95	Nb-95m
Mo-99	Tc-99m
Ru-103	Rh-103m
Ru-106	Rh-106
Pd-103	Rh-103m
Ag-108m	Ag-108
Ag-110m	Ag-110
Cd-109	Ag-109m
Cd-115	In-115m
Cd-115m	In-115m
In-114m	In-114
Sn-113	In-113m
Sb-125	Te-125m

附件六

除役受訓課程講義

EUNDETRAF

Chapter 1

Preparing a decommissioning project

Chapter summary

Introductory Remarks

Decommissioning is the final phase in the life-cycle of a nuclear facilities, i.e. after siting, design, construction, commissioning and operations. Decommissioning refers to the administrative and technical actions taken to remove, in part or in total, the facility from regulatory control. It is a complex process and requires activities such as decontamination, dismantling of equipment and demolition of structures and management of the resulting radioactive wastes. The objective is to progressively reduce the radioactive inventory of the facility in a controlled manner hence reducing the hazard while taking account of the health and safety of the operating personnel, the general public and preservation of the environment. The ultimate goal of decommissioning is unrestricted release or reuse of the site.

The timescale for decommissioning varies according to a variety of factors and can typically range from several years for immediate dismantling to many decades in the case of deferred dismantling. Hence, decommissioning can be a continuous process on a relatively short timescale or a protracted stepwise process involving phases of dismantling followed by periods of storage under institutional control for the remaining radioactive components. The approach to be taken depends on many factors such as:

- National policy and regulation of nuclear activities;
- Availability of waste routes for spent fuel, operational waste and dismantled material;
- Safety of operators and the general public;
- Protection of the environment;
- Structural deterioration of the facility if dismantling is deferred with the attendant problems of surveillance and maintenance;
- Availability of appropriate skills within the workforce;
- Development of cost estimates for decommissioning and the availability of funding to carry out the work;
- Public relation issues;
- Social aspects notably the desirability of providing continuity of employment.

1.1 Fundamentals

Although all decommissioning projects to a certain extent are project and site specific, the planning and licensing documents needed are similar. The difference lies in the degree of detail and the time when they have to be prepared. E.g. in case of a SE-period, detailed dismantling planning documents are prepared at a later time than it is the case for direct dismantling. Thus, the issues to be addressed are the same, since the ultimate objective is the same.

It is also clear, that in case of more modern plants, where a conceptual decommissioning project is part of the documents needed for an operation licence, a certain base has been formed far in advance of the actual decommissioning activities. This plan is site specific and consequently in certain aspects more detailed than decommissioning projects used as basis for the funding calculations, which are normally generic for a certain country.

1.1.1 Basic considerations

There are certain major issues, which must be clarified when preparing a decommissioning project. In the normal case of a planned shutdown of the facility, these issues are clarified well before starting the decommissioning project. These issues can be summarized as:

- Decommissioning strategy
- Financing
- Technical
- Waste Management
- Fuel Management
- Social

Obviously it is necessary to have a secured financing before commencing the project. Furthermore, the basic strategy must be clear, i.e. direct dismantling or dismantling after a safe enclosure period. The techniques to be applied must be readily available or in a defined stage of development. The other technical main issues are the management of fuel, operational waste and dismantled material, i.e. as far as possible the back-end must be available, either as disposal possibility or as interim storage.

As far as possible the licensing environment should be clear, i.e. necessary laws and decrees should be defined. At the same time it must be pointed out that the legal framework should only cover basic aspects like radiation protection, free release limits etc., in order to have some freedom in the execution of the licence.

1.1.2 Boundary conditions

The aspects and issues mentioned above can sensibly be structured in the following way:

• political	⇒	- acceptance by authority and public - legal/licensing constrictions
• technical	⇒	- plant and site condition - plant design - site reuse - availability fuel storage - waste management possibilities and disposal options
• financial	⇒	- availability of budget - cash flow - integration in overall country system
• social	⇒	- personnel age and competence - personnel strategy - integration in project - privatisation strategy.

1.1.3 Project objectives

The overall objectives for a decommissioning project does not deviate extensively from any other project and can be stated as:

- implementation of project
- minimum risk
- as cheap as possible (i.e. generally also as fast as possible)
- socially acceptable.

Based on these basic (partly juridical/legal criteria) objectives the following derived criteria can be listed:

- fulfilment of safety criteria
- minimum costs
- maximum use of own personnel and local companies.
- site reuse
- know-how transfer
- privatisation
- integration in society.

It is clear that for a specific project and a specific site other priorities can be necessary.

Safety criteria

For the overall project the safety objectives can be summarized as:

- guarantee nuclear safety
  - undercriticality
  - cooling
- guarantee appropriate radiation protection
  - limit dose commitment (ALARA)
- conventional workers safety
- release of radioactivity below licensed levels

The first point will obviously not apply, once all fuel has been removed, The remaining three objectives will however be valid until the project end, i.e. the release of the plant from regulatory control or new licence.

1.2 Project analysis

Before starting a project, it is necessary to perform an analysis of the overall project, i.e. its objectives, boundary conditions, risks, economical viability etc., i.e. a so called project analysis.

The objectives of the project analysis can be summarized as:

- evaluation of the project under all boundary conditions, considering feasible alternatives
- determination of main deficiencies and necessary investments
- determination of main conditions and basic time schedules
- determination of requirements on personnel (qualification and number).

The main issues to be covered during the project analysis are typical in the case of decommissioning:

- spent fuel management (and fresh fuel if present)
- waste management
- dismantling strategy
- mass flow logistic
- post operation
- personnel strategy
- site reuse options.

The result of the analysis can be summarized as:

- define main dependencies between activities
- define milestones and overall project life time
- first key decision plan (with arguments).

This information constitutes the basis for the further planning:

Example

To illustrate one of the major results of the project analysis, the key decisions taken in the Greifswald decommissioning project were:

- *direct dismantling instead of safe enclosure*
- *necessity of a new dry spent fuel storage*
- *necessity of an interim storage with treatment capabilities for radioactive material from dismantling due to lack of final disposal capacities*
- *licence for decommissioning and dismantling instead of operation licence prolongation preferable*
- *project realisation by own staff instead of contractors*
- *industrial site reuse instead of green field.*

As can be understood, these are really major decisions, influencing or rather determining all further project activities.

1.3 Technical concept

The technical concept is a rather detailed overview of all aspects of the project. Basically the same issues as in the decommissioning plan are covered, but not to the same degree of detail. A typical table of content is given in **annex 1.1**. This technical concept is the 2nd pillar, 1st is the project analysis, for all further work.

Below some specific aspects of the technical concept will be discussed.

1.3.1 Plant and site conditions

An important aspect for the planning, execution and eventual reuse of site and/or facilities, is the actual state of site and plant. Important here is the physical status of the plant and the history, i.e. notably if incidents or anomalies have taken place and their consequences.

1.3.2 Inventory

Irrespective of whether decommissioning will entail a Safe Enclosure (SE) phase or not, the basis approach is the same:

- An inventory of the plant establishes the physical basis for all planning work.
- Systematizing the planned work facilitates the work and also the discussions with the authorities.

Plant inventory

The inventory will entail the following basic data for the decommissioning planning:

- Systems and other components on a room-by-room basis
- Masses and material types (on a room- and system-basis)
- Contamination in rooms and systems (including estimation of contamination penetration in concrete structures)
- Dose rates, including ambient rates in rooms, hot spots, and at large components
- Hazardous materials (Asbestos, PCB etc.).

These data are the basis for all planning works, especially the dismantling and the management of the dismantled material. The inventory is to a certain extent a living document, due to two reasons:

- Depending on the actual project phase and its need, the degree of detail will be different and will in general increase up to the execution planning;
- Certain important information can only be obtained after shut down and emptying of systems, for example, interior contamination of systems for which sampling is necessary.

Classification of systems

For a systematic planning and licensing approach, all systems are classified according to three categories:

I Systems necessary for nuclear safety and radiation protection

II Systems necessary for industrial safety and operation

III Unnecessary systems

A differentiation is made between category I and II, since the requirements on the systems are different due to their function. The systems in category III can be dismantled after separation from the remaining systems.

It is also clear that the categorization of a system can change with the project phases, but also within a project phase. E.g. when all nuclear fuel has been removed, all related nuclear safety systems will be changed from category I to category III.

Classification of material (radiological)

Category I	unrestricted material from the monitored area, which is not subject to the release measurement procedure
Category II	suspected material (an eventual contamination cannot be excluded) from the monitored area, mainly from the turbine hall
Category III	contaminated material

Classification of systems/components (dismantling)

- Kinds of material
- Geometries
- Rooms
- Disposal routes

1.3.3 Waste and material management

During the decommissioning of a nuclear power plant, large amounts of material have to be handled and treated in a short time. This means that for a successful project a detailed inventory and careful planning have to be performed in order to achieve an optimal mass flow in the facility and on-site. A primary objective in this context is to install techniques and procedures for free classing of material, reuse of material and use of decay storage. The introduced waste management scheme must guarantee a logistic, with ample buffer storages to avoid bottlenecks as well as handling and treatment systems minimising the amount of radioactive waste. In this way, the basis is laid for a cost effective project.

The waste management concept of EWN is e.g. based on the following boundary conditions and principles:

1. Provision of sufficient buffer and intermediate storage capacities to achieve a high flexibility in the logistics and waste management. Therefore, the construction of the Interim Storage North (ISN) was of essential importance.

2. Removal of the spent fuel from the reactors and cooling ponds to the wet interim storage and later transport in dry CASTOR cask to the ISN.

3. Installation of equipment for the treatment of dismantled material using modern technologies for the reduction of dose exposure and increase of efficiency.

4. Further use of the existing waste facilities, upgrading or extension, as far as it is economically justified.

5. Use of the limited storage capacity of the ERAM (salt-mine) until 2000 as far as possible (closed 1998).

6. Permanent optimisation of treatment technologies and of logistic to increase efficiency and to avoid bottlenecks (e.g. installation of second free release facility in 2002).

1.3.4 Plant adaptations

Plant adaptations are mainly necessary due to two different aspects:

- adaptation (decrease) of certain system capacities in order to increase cost efficiency and
- adaptation due to new legal requirements.

1.3.5 Project structuring

The project is timely limited and with defined content and goals and it is not repeatable. Thus, a vast amount of activities have to be executed, which all are unique and timely limited. To order all these activities a hierarchic structure is necessary. Different structures can be created depending on from which point of view one looks at the project. In this way the following structures are applicable:

- Work breakdown
- Responsibility
- Objects
- Type of work
- Phases.

The primary structure is the work breakdown structure. The other structures are used when the corresponding specifications have to be evaluated or studied. The project structure must be used by all participants and it may in general not change during the project[1]. In this way it guarantees law and order in the project work. It allows in this framework also the structure needed for the IT databank.

Work breakdown structure

The work breakdown structure is directly related to the works to be performed in the project. This relational project structure is a mixed structure since it contains basic elements from other structures, e.g. phases (planning, execution...), works (electrical, mechanical...) or objects (turbine hall, waste building...). This is used as the primary project structure and the following hierarchic levels are used.

[1] If for some reason a change must be performed a special procedure must be applied.

- Megaproject
- Project
- Part project
- Program
- Working package
- Activity
- Action
- Task

Each level is the sum of the level below and thus a concentration of the information takes place.

Phases

After the operation phase a number of phases can be envisaged:

- Shut down
- Post Operation
- Preparation of Safe Enclosure (SE)
- SE operation
- Remainder system operation
- Deferred Dismantling
- Dismantling
- Site Clean up

The actual phasing is project specific and will be worked out during preparation of the Final Decommissioning.

1.4 Decommissioning plan

A decommissioning plan (DP) should cover all aspects and issues of relevance for a certain decommissioning project. The degree of detail will increase the closer to execution the project is. Obviously, the details of the DP are site specific.

1.4.1 Content and structure

The decommissioning plan requires competence within the following areas:

- Comprehensive Project Management,
- Licensing Process,
- Characterisation of the facility (dose mapping and inventory classification)
- Safety assessment,
- Criticality assessment,
- Radiation Protection and Monitoring,

- Environmental impact assessment,
- Waste Management,
- Decontamination,
- Dismantling and demolition,
- Remote dismantling,
- Engineering support,
- Quality Assurance and Quality control,
- Physical Protection,
- Site adaptation and development,
- Public relation

A detailed (final) Decommissioning Plan [1.1] should cover the following major issues:

- a description of the nuclear reactor (or facility), the site and the surrounding area that could affect, and be affected by, decommissioning;

- the history of the nuclear reactor, reasons for taking it out of service, and the planned use of the nuclear installation and the site during and after decommissioning;

- a description of the legal and regulatory framework within which decommissioning will be carried out;

- explicit requirements for appropriate radiological criteria for guiding decommissioning;

- a description of the proposed decommissioning activities, including a time schedule;

- the rationale for the preferred decommissioning option (if selected);

- safety assessments and environmental impact assessments, including the radiological and non-radiological hazards to workers, the public and the environment; this will include a description of the proposed radiation protection procedures to be used during decommissioning;

- a description of the proposed environmental monitoring programme to be implemented during decommissioning;

- a description of the experience, resources, responsibilities and structure of the decommissioning organization, including the technical qualification/skills of the staff;

- an assessment of the availability of special services, engineering and decommissioning techniques required, including any decontamination, dismantling and cutting technology as well as remotely operated equipment needed to complete the decommissioning safely;

- a description of the quality assurance programme;
- an assessment of the amount, type and location of residual radioactive and hazardous non-radioactive materials in the nuclear installation, including calculation methods and measurements used to determine the inventory of each;
- a description of the waste management practices, including items such as:
  - identification and characterization of sources, types and volumes of waste;
  - criteria for segregating materials;
  - proposed treatment, conditioning, transport, storage and disposal methods;
  - the potential to reuse and recycle materials, and related criteria; and
  - anticipated discharges of radioactive and hazardous non-radioactive materials to the environment;
- a description of other applicable important technical and administrative considerations such as safeguards, physical security arrangements and details of emergency preparedness;
- a description of the monitoring programme, equipment and methods to be used to verify that the site will comply with the release criteria;
- details of the estimated cost of decommissioning, including waste management, and the source of funds required to carry out the work; and
- a provision for performing a final confirmatory radiological survey at the end of the decommissioning project.

The Plan is a document (or a set of documents) that in practice usually will be adapted over time as more detailed or new information becomes available. Because, the plan is normally a matter of licensing approval, it should be issued flexible enough to be able to implement changes later on in the project without the necessity for a new formal application procedure.

Practical Example

Within the NPP Greifswald Decommissioning Project, the original Decommissioning Plan presents a set of 15 documents, here called "Explanatory Report" (ER). In preparation of the first licensing step (phased licensing process), the set of documents was subject to some relevant changes as shown in table below:

ER	First issue in 1994	Issue Index one year later
ER1	Licensing concept for decommissioning/dismantling of units 1 - 6	2
ER2	Radiation protection concept	2
ER3	Site operation	4
ER4	Incident analyses	2
ER5	Fire protection concept	2
ER6	Intervention in contaminated building structures under controlled area conditions	3
ER7	Reconstruction/new construction of the complex of central active workshop/melting facility	3
ER8	General dismantling plan	1
ER9	Material flow/release measurement concept	1
ER10	Decontamination concept	2
ER11	Disposal of damaged fuel elements	1
ER12	Physical protection	3
ER13	Remote dismantling of reactor pressure vessels 1 to 5	2
ER14	Clearing of tube storages for activated components	1
ER15	Disposal of absorber sections and shielding assemblies	2

The issue indexes one year later were approved within the appraisal of the first (umbrella) licence.

1.4.2 *Initial, on-going and final DP*

Usually, the establishment of a initial decommissioning plan of any nuclear facility is already part of the design documentation. This decommissioning plan is focused on the technical aspects and serves as the very first basis to start decommissioning preparation.

During operation, this plan must be updated regularly (every 3-5 years) depending of various factors like for example regulatory requirements, plant modifications implemented, operational events, technological developments in decommissioning etc.

After the timing of the final shutdown is known and the decommissioning strategy has been defined, a detailed (final) decommissioning plan covering all aspects of decommissioning must be prepared by the decommissioning management team and approved, as appropriate, by the regulatory body.

1.4.3 International approaches

There are certain activities going on, internationally, with the aim to exchange know-how and issue guidance. This work is performed in 3 different forms, i.e.:

- European Community (Framework Programme)
- IAEA
- OECD/NEA

1.5 Licensing aspects

The most important aspect in the licensing procedure is the potential risk in the different decommissioning stages. These risks give the frame to the safety levels **(see figure 1.1)**.

It is clear that the amount of licensing documents has to be limited to a sensible level, i.e. what is really necessary from safety point of view (risk to workers, population and environment). Generally speaking the safety criteria to be applied are determined by:

- the inventory of radioactive materials
- the condition of the facilities
- the existing safety measures.

Thus, it can be concluded, that when the fuel has been removed from a plant, the decommissioning or rather the dismantling activities are comparable with a major revision phase (excluding dismantling of activated components). This fact can also be seen in many countries, where it is actually possible to perform major, if not all, dismantling activities with the operation licence.

In certain projects it can be observed, that the licensing procedure and the regulatory control is not clearly linked to the risks involved. Furthermore, the laws governing industrial activities and environmental issues are a function of the economy in the country in question, i.e. the better the economy the more stringent laws and release limits are introduced. Thus, all stakeholders have to take care, that one does not introduce more severe laws and regulations than is necessary from safety/risk point of view. This will only hamper the development in the country and obviously slow down the project.

With the licensing strategy, the applicant influences
- the duration of the whole decommissioning process,
- the need of personnel and technique,
- the total costs (including waste management) and
- the acceptance of the public.

1.6 Financial aspects

The costs for decommissioning must in the normal case be collected during the operation of the facility. This is e.g. valid for nuclear reactors in most countries where the further decommissioning costs are accumulated from a levy on the electricity price.

1.7 Social aspects

The social issue is in many projects a crucial issue. This is notably the case in the East European plants (previous communist area), where a much larger workforce is present on-site. In order to implement a successful project, it is mandatory to resolve the personnel issue as soon as possible.

In case of EWN the main problem after shut down was to reduce the number of employees, since this under all circumstances was much too high.

To solve this problem, the following measures and principles were introduced:

- No major contractors
- Retirement scheme
- Privatisation/outsourcing
- Education
  - decommissioning
  - better position on labour market
- Dismissal with economical support
- Re-industrialisation of the site.

1.8 References

[1.1] IAEA, "Decommissioning of Nuclear Power Plants and Research Reactors";-Safety Standard Series No. WS-G-2.1, IAEA, Vienna, 1999

Annex 1.1

Content of Technical Concept

1. Introduction
2. Site and buildings description
3. Present status
4. Decommissioning and dismantling activities
5. Decontamination methods
6. Emissions
7. Material flow and waste management
8. Radiological protection
9. Declassification
10. Necessary new installations and systems
11. Fire protection
12. Workers protection
13. Plant security
14. Infrastructure
15. Safety case
16. Environmental impact assessment

1 Introduction

- objectives
- boundary conditions
- strategy and management

2 Site and buildings description

- site layout
- description of buildings
  - design features
  - planned modifications
  - barrier functions
- site during decommissioning

3 Present status

- nuclear fuel
- radioactive waste
- facilities to be put in safe enclosure
- facilities to be decommissioned
- post operation facilities and systems

- contamination and dose rate mapping
- mass and material inventory
- existing cranes and other handling and transport means

4 Decommissioning and dismantling activities

- major components with their characteristic data (size, mass, material, activity)
- necessary new cranes and transport means
- establishment of the safe enclosure
  - limits, boundary conditions
  - constructional modifications
  - necessary systems and facilities
  - dismantling activities
- cutting methods and tools
  - mechanical
  - thermal
  - filtering system
  - manual and remote operation
- transport of dismantled material in plant and on site
  - air locks
  - transport containers
  - cranes and other means

5 Decontamination procedures

- decontamination of systems
  - necessary operation of the systems
  - decontamination agents; treatment and conditioning thereof
  - objective and argumentation
- decontamination of dismantled parts
- decontamination of buildings
- decontamination of site
- secondary waste

6 Emission during normal operation

- release with air
  - determination of releases with air and water, as function of the decommissioning status
  - amount, origin and cleaning of each air stream
  - upper limits of yearly releases of β/γ - aerosols, α -aerosols, I, H-3, C-14...
  - measures to limit short term releases
  - detailed specification of nuclides
  - localisation and height of emissions on site
- release with water
  - as above

7 Material flow and waste management

- disposal/release paths
- production; type, mass, activity, rate, activity limits
- minimization of radioactive waste
  - technical and organisational measures
  - collection, sorting, record
  - proof of reuse dose commitment estimation
  - clearance of material
- treatment and storage of radioactive material (type, amount, activity)
  - pre-treatment; collection, sorting, cutting, decontamination, packing, interim storage, transport
  - conditioning (facilities, external/internal, mobile/fixed, container, production of container types
  - interim storage, on site/external
  - final storage

8 Radiological protection and surveillance

- description of filtering and cleaning systems
- description of installations for storage, treatment and conditioning of radioactive materials
- description of radiological protection measures
- description of off site surveillance
- plan/list of radiation protection areas

9 Declassification

- clearance/declassification limits
- schedule for exemption with conditions for:
  - procedures and methods to reach activity free conditions
  - clearance measurements
  - administrative procedure

10 Necessary new installations and systems

- summary of rules and criteria
- summary descriptions of:
  - function during normal operation and in accident situations
  - simplified flow sheets
  - implantation of main components
  - provisions for accidents
  - connection to other systems
  - activity inventories; measures for containment
  - QA system

  - maintenance
  - instrumentation system
- for electrical systems especially
  - emergency supply
  - protection system, lightening, earthing

11 Fire protection

12 Workers protection

13 Plant security

14 Infrastructure

15 Accidents

- identification of relevant accidents (load, leakage, fire etc.)
- clarification of external accident sources (air plane crash...)
- evaluation of accident scenario
- radiological analysis of the covering accident, i.e. determination of the radiological source:
  - activity content in the plant parts in question
  - activity release as function of time
  - efficiency of filtering equipment
  - evaluation of possible secondary effects

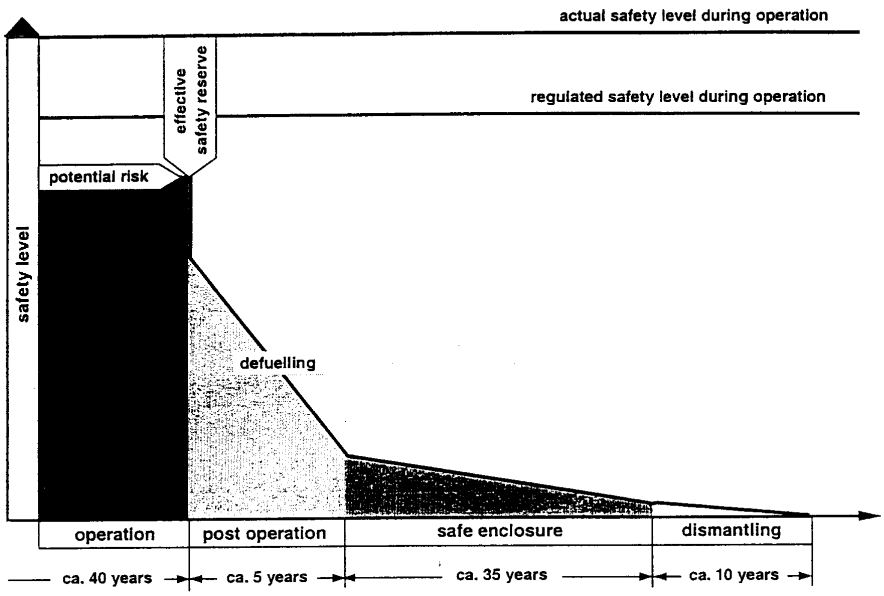


Figure 1.1 Decrease in potential risk

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Chapter 2

Nuclear and non-nuclear safety aspects

Chapter summary

Nuclear and non-nuclear safety aspects

2.1. Nuclear Safety

- The safety of nuclear installations

The main safety objectives as far as nuclear installations are concerned may be presented like they are mentioned in reference 2.1.

- General Nuclear Safety Objective

"To protect individuals, society and the environment form harm by establishing and maintaining in nuclear installation effective defence against radiological hazards." This objective is supported by two complementary objectives: the technical aspects related to administrative and procedural measures ensure defence against hazards due to ionizing radiation.

- Radiation Protection Objective

"To ensure that in all operational states radiation exposure within the installation or due to any planned release of radioactive material from the installation is kept below prescribed limits and As Low As Reasonably Achievable (alara), and to ensure mitigation of the radiological consequences of any accidents."

- Technical Safety Objective

"To take all reasonably practical measures to prevent accidents in nuclear installations and to mitigate their consequences should they occur; to ensure with a high level of confidence that, for all possible accidents taken into account in the design of the installation, including those of very low probability, any radiological consequences would be minor and below prescribed limits; and to ensure that the likelihood of accidents with serious radiological consequences are extremely low."

In order to achieve these three objectives, a safety analysis has to be performed. This analysis has to be done during the design phase but also during the lifeperiod of the plant. Furthermore, in case of incident or accident, an analysis has also to be done to identify the origin of these events and to provide guidance or to initiate remediation actions. For instance, the analysis before the plant is in operation will examine:

- all planned normal operational modes of the plant;
- plant performance in anticipated operational occurrences;
- design basis accidents (such as loca for example);
- event sequences that may lead to severe accidents.

It is worthwhile to note that not only the internal but also the external sources of hazards have to be identified and taken into account in such analysis.

- The concept of defence in depth

This concept is a very important one. It deals mainly with the origin phase of a nuclear installation. How this concept is applied depends also upon the nature of the installation. Nevertheless, the general principles as recalled here remain valuable.

There are five "levels of defence":

level one

The aim is to prevent deviation from normal operation and to prevent system failures (design, construction, maintenance and operation) based on the application of redundancy, independence and diversity. Procedures should be available for each period of life of for each operation needed (maintenance, repair, testing for example).

level two

The aim is to detect and intercept deviation from normal operational states in order to prevent anticipated operational occurrences from escalating to accident conditions.

level three

Here it is assumed that the escalation of certain anticipated operational occurrence or postulated initiating events (pie) may not be arrested by a proceeding level and a more serious event may develop. Inherent safety features, fail safe design, additional equipment and procedures are provided to control their consequences and to achieve stable and acceptable plant states following such events.

level four

The aim is to address severe accidents and to ensure that radioactive releases are kept as low as practicable (protection of the confinement function).

level five

The aim is to mitigate the radiological consequences of potential release of radioactive materials that may result from accident conditions. Emergency preparedness is an example of what should be available in such circumstances.

Without providing too many details, it is straightforward that in order to achieve these safety objectives and to comply with the five levels of defence, all the partners have to be involved (management, workplaces...).

- Safety assessment

As already mentioned, the assessment of the safety has to be performed in order to achieve the safety objectives. Such assessment will make use of methods like Hazard and operability study (hazop), Failure Modes and Effects Analysis (fmea).

The fmea is illustrated in **figure 2.1**.

Item	Nr.	Way of failure	Cause of failure	Effect – Conseq.	Risk-assessment	Measures	Sequence
safetyvalve	1	continuous open	blocked	loss compressed air	10	daily visual check	2
	2	continuous open	main broken	loss compressed air	0,2		3
	3	continuous closed	stuck	overpressure	200		1
	4	partly opened	blocked	loss compressed air			
compressor	5	doesn't work	no voltage			emergency feeding	
	6	doesn't work	broken crankshaft				
	7	doesn't work	valve failure				
decrease compressed air buffer cask							
manometer							
switch							

Other examples may be found in the literature concerning risk assessment (references 2.2. – 2.3.). Most of these methods require the evaluation of the risk. To do this, it's interesting to recall that a risk can be defined as:

$$R = P \times G$$

Where R = risk, P = probability of occurrence and G = gravity of the effects.

The estimation of both quantities, P and G, are very important. The probability of failure of a component can be obtained using data provided by the manufacturers. Sometimes, one needs to work with assumptions. But, one of the most important results of the accident analysis for instance is that such estimation of risks may take advantage of the data that are then collected. So the role of such analysis is of a prior importance.

It's also worthwhile to mention that safety assessment is an unceasing process. International assessment is strongly recommended.

- Nuclear Safety and Decommissioning

The Safety Objectives and the way to achieve these objectives have to be put into practice in a lot of different installations and during various stages of the lifetime of these installations.

Let us now try to focus on the decommissioning of Nuclear Power Plants (NPP). The general considerations we mentioned here above have to be "translated" and, furthermore, special facts have to be dealt with.

As a general statement, the following topics have to be stressed out as far as decommissioning is concerned:

- Knowledge of the installation:

Decommissioning has to be considered as a yard. This means that the works taking place are quite similar to the building of the plant. Furthermore, there are always discrepancies between the plan of the installation and the installation itself. Indeed, some modifications to the plant have been brought during its life and they haven't frequently been taken into account for updating the description sheets of the plant. So "surprises have to be expected!"

- Nature of the operations:

During the decommissioning of a NPP, the works themselves are quite unusual and may never been considered as "routine" conditions. The environment of the workers is almost continuously changing.

- Nature of the tools:

The operation concerning the dismantling requires new tools. These are very often new tools for the operators but also, some tools have been especially developed for such operations.

- Behaviour of the manpower:

As far as the workprocedures and the daily approach are concerned, dismantling brings very significant changes. As already written, the "human environment" is quite seriously modified.

- State of the Regulations:

Decommissioning may, in some countries, lead to some problems concerning the regulations. Consequences of a decommissioning such as the management of nuclear waste, the free release of waste,... are examples of topics which require a legal framework. This waste is not always available or is still in progress.

So, the management of radiological risks (see also § 2.2.), have to be adapted and the general rules as already mentioned have to be extended.

The main problems or issues faced when dealing with decommissioning are probably the following ones:

- required presence in areas with potentially high radiation field;
- opening of loops and piping with internal contamination;
- continuously changing environment by the dismantlement of parts of the installation, sometimes including shielding;
- potential unforeseen situation due to the quality of the remaining documentation and to the fact that the installations were often never foreseen to be decommissioned;
- "one-shot" operations, implying sometimes heavy preparation to be distributed only to one operation, thus economically less interesting than repetitive operations.

All these situations were encountered in the BR3 pilot decommissioning project, and led to the implementation of a formal ALARA procedure to deal with such kind of issues.

The BR3 (for Belgian Reactor Nr 3) was the first PWR installed in Europe. In service since 1962, it was shutdown in 1987 after 25 years of operation. It is a quite small reactor with a thermal power of 40.5 MW_{th}. At the end of its operating life, the European Commission, in the framework of its five-year plan of RTD, selected in 1989 the BR3 as one of the four pilot decommissioning projects on decommissioning of nuclear installations.

The main steps of the decommissioning programme up to now were:

- Full System Decontamination of the primary loop in 1991;
- dismantling of the high active thermal shield with three different techniques, of the Vulcain internals and of the first set of Westinghouse internals (30 years decay time) by mechanical cutting up to 1996;
- the dismantling of the auxiliary circuits started in 1995 and is still going on;

- the dismantling of the reactor pressure vessel ended in 2000;
- the construction and start of the exploitation of thorough decontamination processes for dismantled pieces in the period 1996 to 1999.

For the dismantling of the contaminated loops, we decided to reduce the dose rate of the loops by chemical decontamination, to dismantle the pieces and to sort them following their specific evacuation route and, finally, to thoroughly decontaminate the pieces with the objective to minimise the amount of materials disposed of as radioactive waste.

Moreover, the clean up of the whole research centre, financed by a technical liabilities funding set up by the Belgian Government in 1989, involved also the dismantling and decommissioning of hot cells and contaminated research laboratories, some of them having even led to the official release for unrestricted reuse of the building.

More than 10 years of practical experience on decommissioning has thus been accumulated through these projects. The BR3 decommissioning project was not only selected as European pilot project but was also considered as Belgian pilot for the Authorities and as SCK•CEN pilot project for the complete implementation of the ALARA principle and for an integrated approach of industrial and radiological safety.

But the optimization of the radioprotection should take complete operations into consideration, in order to avoid transferring the radiological risk from one operation to the next one or to a future operation in the follow up of the project. Let us illustrate this by two examples.

The first one concerns the decontamination of the primary loop and auxiliary circuits. This operation concerned the whole primary loop, including the reactor pressure vessel. The operation itself implied a significant dose uptake for the operators, mostly because the reactor head had to be closed, and this activity is still performed "by hands" on this old reactor generation. The total dose uptake for the operation was about 160 man-mSv (preparation and post decontamination operations included), but allowed to save between 4 to 7 man-Sv for the future dismantling of the primary loop (carried out almost 10 years later). The dose rate in the vicinity of the loop was reduced by a factor of at least 10, leading to an ambient dose rate between 60 and 80 μSv/h.

This shows that the optimisation of the radioprotection must take future operations into account for getting the whole scene.

The second one concerns the optimisation of the dismantling of the primary loop piping and all auxiliary equipments situated in the primary loop area. This operation was quite complex, involved a lot of equipments and components, and let various potential alternatives open for the chronology and working procedure.

The SCK•CEN developed at the end of the 90's a user-friendly optimisation tool (so called "Visiplan") which allows to analyse rapidly, and in a 3D graphical way, different scenarios and procedures. The area was then simulated (**see Fig. 2.2.**), and different alternatives were envisaged. The use of this efficient tool allowed to optimise the procedure and to take into account the changing environment of the plant.

A schematic representation of the operation room is given in **fig. 2.2**

Some "driving forces" for such working circumstances may be indicated. They are the result of about twelve years experience in the decommissioning of a small NPP. These driving forces may be listed as follows:

- open and frequent communication;
- support from the Health Physics and Safety Department of the site;
- development of particularly adapted tools.

Furthermore, one needs also to manage the dismantling yard with more flexibility and more open-mindedness. For instance, authorisation for the operations should be delivered on a step by step process. This means that "cold tests" and/or "hot tests" will be allowed. Use of mock-up will be strongly supported by the Health Physics and Safety Department.

As far as communication is concerned, one should be taken to involve all the partners. Meetings with the workforce, with the Health Physics and Safety Department (and with the Regulatory Body if needed) are strongly recommended. Furthermore attention should be paid to the training of the workers. External workforce will be involved in order to be aware of the rules for preventing radiological (and non-radiological) risks.

"Last but not least", the ALARA approach may be considered as a very efficient way to manage such dismantling yard. Indeed, as already mentioned (reference 2.4.), ALARA principle is not only a "tool" but has to be considered as a mean to develop the safety culture in such environment.

2.2. Nuclear Radioprotection

- Irradiation en contamination risks
- Analysis of the work environment - ALARA : Management of the doses
- Controls of the dismantling operations

2.2.1. *Irradiation en contamination risks*

2.2.1.1. Introduction

In a nuclear environment, we can encounter different types of radiation. These are summarised in the table below.

	Type	Charge	Mass
α	Particle $_2^4He$	+ 2	+ 4
β	Particle e^- of e^+	- 1 of +1	1/2000
γ	Wave	0	0
neutron	Particle	0	1

For the purpose of effective protection against the hazards of ionising radiation, it is very important to know the difference between irradiation and contamination.

Upon irradiation, the radiation comes from outside the body, for example through the wall of a waste container or from a pipe containing a radioactive liquid. In principle, it is known where the substance is located, there is no risk for direct contact with it, and especially the radiation with a large penetration power (γ radiation, for example) is important.

Upon contamination, there is a spreading of the radioactive material, and there is a risk of direct contact with it (for example: a puddle under a waste container, radioactive dust,...). In this case, there is a risk of contamination of the skin and of intake to the body.

The most important differences in practice are summarised in the following table:

	irradiation	internal contamination
definition	source always outside the body	source in the body
most dangerous	γ, n radiation with high penetration power (through clothes, skin, ...) :	especially α energy deposition of the radiation occurs in a small area of the tissue
exposure time	time of being in the vicinity of the source	time of the presence of the isotopes in the body (activity disappears through decay or elimination)
control and registration	easy systematically applicable	strongly depending on the isotopes

The characteristics of the different types of radiation in the air are summarised in **Fig. 2.3.**

As we can see the more dangerous radiations for an external irradiation are the gammas and neutrons.

2.2.1.2. Definitions

Different terms are used to characterise the impact of the radiations on the human body.

- Absorbed dose D

The absorbed dose D can be defined by the energy absorbed per unit mass. In the context of radiation protection the absorbed dose denotes the dose averaged over a tissue or an organ. The unit for absorbed dose is the Gray (1 Gy = 1 Joule/kg). The former unit was the rad (1 Gy = 100 rad).

- Equivalent dose H_T

The absorbed dose, in tissue or organ T weighted for the type and quality of radiation R. It is given by:

$$H_{T,R} = W_R D_{T,R}$$

Where

$D_{T,R}$ is the absorbed dose averaged over tissue or organ T, due to radiation R,
W_R is the radiation weighting factor.

When the radiation field is composed of types and energies with different values of W_R, the total equivalent dose, H_T, is given by:

$$H_T = \sum W_R D_{T,R}$$

The appropriate W_R values are specified in the table below

Type and energy range	Radiation weighting factor, W_R
Photons, all energies	1
Electrons and muons, all energies	1
Neutrons, energy < 10 keV	5
10 keV to 100 keV	10
> 100 keV to 2 MeV	20
> 2 MeV to 20 MeV	10
> 20 MeV	5
Protons, other than recoil protons, energy > 2 MeV	5
Alpha particles, fission fragments, heavy nuclei	20

The unit for equivalent dose is the Sievert.

- Effective Dose

The effective dose can be defined as the sum of the weighted equivalent doses in all the tissues and organs of the body from internal and external irradiation. It is defined by the expression:

$$E = \sum W_T H_T = \sum W_T \sum W_R D_{T,R}$$

Where

$D_{T,R}$ is the absorbed dose averaged over tissue or organ T, due to radiation R,

W_R is the radiation weighting factor and
W_T is the tissue weighting factor for tissue or organ T.

Values of tissue weighting factor, W_T, are shown below:

Tissue or organ	Tissue weighting factors, W_T
Gonads	0,20
Bone marrow (red)	0,12
Colon	0,12
Lung	0,12
Stomach	0,12
Bladder	0,05
Breast	0,05
Liver	0,05
Oesophagus	0,05
Thyroid	0,05
Skin	0,01
Bone surface	0,01
Remainder	0,05

The unit for effective dose is the Sievert.

- Collective Dose

The collective dose (S) of a population or a group exposed to a source or a practice with irradiation as a consequence, is given by the following equation:

$$S = \sum H_i \cdot P_i$$

wherein H_i represents the average of the dose received and to be expected, at the level of the whole body, of an organ or a tissue, by the P_i members of the i-th subgroup of the population or group.

Unit man-Sievert (man.Sv)

- Dose limitation

The European directive Euratom 96/29/Euratom has defined a dose limitation for the workers of the Member States:

The limit on effective dose for exposed workers shall be 100 millisieverts ('mSv') in a consecutive five-year period, subject to a maximum effective dose of 50 mSv in any single year. Member States may decide an annual amount.
Without prejudice to paragraph 1:
(a) the limit on equivalent dose for the lens of the eye shall be 150 mSv in a year;
(b) the limit on equivalent dose for the skin shall be 500 mSv in a year.

In many national legislations, this limitation has been interpreted as 20 mSv/year.

2.2.2. *Analysis of the work environment - ALARA: Management of the doses*

The best way to optimise the protection against the risk of harmful effects of radiation is to keep the dose incurred "As Low As Reasonably Achievable" (ALARA).

2.2.2.1. Protection against irradiation

The dose incurred may be limited by:

- decreasing the exposure time;
- increasing the distance;
- limiting the effective force of the source, possibly by application of the appropriate shielding.

See Fig. 2.4

A few practical examples:

- Do not remain in the vicinity of a radiation source if it is not absolutely necessary; during the supervision of a work, or when waiting until colleagues have completed a job, one should try to remain as far as possible from known sources.
- Do not sit on containers (not even if they are tightly closed), drums or pipings with radioactive material (small distance, large exposure time, no use).
- Use tongs to move a source instead of your bare hand (if the source is not to big).
- Do not use excessively strong sources, for example for the energy calibrations of a detector.
- Reduce the source term by adequate shielding, if the exposure due to the installation of the shielding is smaller than the exposure incurred when carrying out the job without the shielding; reduce the exposure time by adequate training with inactive "dummies".

2.2.2.2. Protection against intake to the body

The main pathways of intake to the body are:

- through inhalation (i.e. breathing)
- through ingestion (i.e. swallowing)
- through contamination of wounds
- through uptake by the skin (in the case of certain isotopes or substances).

The protection against this may be ensured by:

- Avoiding contamination of workplaces, air, ... and regular checks of material for work...;
- Clearly delineating and indicating contaminated areas;
- Wearing appropriate protective clothing in controlled areas (overshoes, gloves, plastic overalls...) and showing the required discipline when leaving them

(putting away contaminated or possibly contaminated clothing separately, controlling used material...);
- Refraining from eating, smoking, drinking in controlled areas; not introducing products to eat, drink or smoke in areas or laboratories with contamination risks.

Against intake through breathing:

- good ventilation (exhaust booths);
- regular checks of air contamination in potentially dangerous locations;
- wearing of gas masks with combined filters (absolute filter + active carbon part).

Against intake through ingestion:

- wearing of gloves, and their regular replacement when they are contaminated;
- use of glove boxes...

Against contamination through wounds:

- in case of wounds, always seek medical advice, certainly when the wound was incurred in a possibly contaminated working environment.

Against uptake through the skin: working in plastic overall.

2.2.2.3. ALARA-analysis of a work.

Before starting dismantling operations, an ALARA approach has to be defined. The first step in decision making is a clear recognition of the problem, that's to say what has to be dismantled, what are the initial radiological conditions of the work environment (doserate, contamination values,...).

The following step consists in studying different options for dealing with the problem. To identify options available and when considering a problem afresh, it will often be useful to return to the simplest formulation of the collective dose equation:

$$S = d.t.N$$

Where S = Collective dose
d = Average dose rate
t = time of exposure (i.e. duration of the task)
N = number of people exposed.

Radiation protection options will work upon one or more of the three components of this equation, and will be chosen with the aim of reducing collective dose and/or the probability of exposure, making allowance for the distribution of individual doses.

Nevertheless, the ALARA-planning is a difficult task especially when multiple radioactive sources are present in the work environment. This task becomes even more difficult when the environment has a complex geometry. That's the reason why it becomes interesting to simulate the work environment in 3D in software, such has VISIPLAN, based on information about the geometry, the materials and the radiation field.

The VISIPLAN methodology is characterised by three different stages:

- The information gathering and model building stage
- The general analysis stage
- The detailed analysis stage and the work planning
- The follow-up stage.

In the first stage the computer model of the environment is built based on the known geometry, the materials information and information about the radiation sources of the site. When the sources are known a calculation of the field can be performed immediately. When no information about the source intensities are present a source inference algorithm provides the possibility to determine source strenghs from a semi-detailed dose mapping of the working area. The mapping together with the information about the history of the site results in pinpointing the position of the main sources that contribute to the field. The end of this stage results in the basic geometry from which other geometries, mostly with supplementary shielding, are derived.

In the general analysis stage the calculated field is studied and suggestion about shielding techniques are tested and analysed using calculated dose map for each of the suggested shielding geometries.

Once a shielding geometry is chosen, a detailed dose calculation can be performed along a trajectory which is constituted of a series of tasks each characterised by a position, a task description and a work duration. Several trajectories can be calculated in different shielding geometries.

A set of scenarios can then be build from a selection of trajectories in the different geometries. The intercomparison of these scenarios then leads to an optimisation of the work to be performed.

In the follow-up stage the dose accounts of the workers are compared with the prediction from the model. When large deviations occur, a reassessment of the work can be performed by adapting the model to the new information. This makes it possible to adjust and thus to further optimise the work during its progress.

Let take the example of the dismantling of a Hot Cell.

Before starting we will thus carry out a radiological assessment of the work environment. This radiological assessment will be first done by means of doserate measurements inside the cell. At this time we will for example measure global doserate from 1 to 6 mSv/h, with hot spots up to 20 mSv/h. Smear samples can also be taken and afterward measured in gammaspectrometry. For example, it will permit to identify elements such as ^{60}Co and ^{137}Cs.

Based on the above information and information about the dimensions of the Hot Cell the model can be build (**see Fig. 2.5**).

Calculations are afterwards carried out to have an idea of the different doserates inside the cell in function of the different positions and to see if it's possible to reduce these, using for example shieldings of working at distance (**see Fig.2.6**).

When the basic radiological information is well known, it's time to divide the work in different trajectories to realise it, each composed of a series of tasks. For the dismantling of a Hot Cell, these tasks can be the installation of materials, the cutting of the working table of tanks,...

On the base of the different calculated trajectories, a scenario can then be defined and optimimized, to reduce the doses to a maximum. At that time a collective dose for all the operations will also be calculated. In or example, for such a dismantling operation, a collective dose of about 25 man.mSv can be foreseen.

2.2.3. *Controls of the dismantling operations*

Before starting the dismantling operations, all the information about how they will be carried out has to be gathered in a procedure that has to be approved by the Safety Department. This procedure will not only describe the operation that will be carried out, but also the protective measures that have to be taken to protect the worker and the work environment.

2.2.3.1. Protection of the work environment

When dismantling all nuclear facilities, the worker will be first confronted with contamination of the surfaces to cut. This contamination will then have the possibility to come in suspension in the air when these elements are manipulated. There is thus a risk of internal contamination of the workers.

The first way to reduce this risk is to isolate the working environment of the rest of the zone. This can be done by means of ventilation. A depression has to be created inside the working environment (more air is evacuated that air is introduced).

At this stage we can be confronted with different types of working environment. For example all the rooms of zone can be considered as the working environment and the depression inside this room will thus be greater in comparison with the neighbouring rooms. If the working environment is only a part of the ventilated room, a special zone with local ventilation will have to be foreseen. This zone can be composed of a plastic tent connected to a local group of ventilation. In function of the contamination risk, the evacuated air from this tent will be filtrated and directed to the exhaust of ventilation system of the installation.

2.2.3.2. Protection of the worker

a) Air contamination

Confronted with a risk of air contamination, the worker will also have to be protected. In function of the nature and importance of the contamination, he will use a masker with a P3-cartridge (against radioactive particles), a combined cartridge (against radioactive particles and iodine) or an overpressure pack with or without mask.

The choice between the different personal protective means against air contamination is of the responsibility of the Radioprotection Officer. For example, an overpressure pack with mask will be used in an environment highly contaminated with alpha-particles (**See Fig. 2.7 and 2.8**).

b) Surface contamination

In function of the contamination risk, the worker will wear specific clothes for the control area, such as an overall, gloves, overshoes. He will change these clothes when he will leave the control area.

Before leaving the zone, the worker will have to undergo a control of contamination either by means of a "port-monitor", that allows to control the whole body for contamination or a Hand&Foot monitor.

c) Internal contamination

In case of suspicion of internal contamination different types of measurement can be carried out. For γ-emitters a measure at a Whole-Body Counter can be carried out. In this case, the γ's from inside the body are measured by sensitive detectors. The total activity inside the body of the different elements can then be determined.

For α-emitters, a direct measurement is impossible, due to the short distance of the particles in matter. A nose-blow has first to be taken and then analysed in α-total. In function of an intervention level (for example 0,5 Bq), different products will be given to try to accelerate the evacuation of the activity in the body. Different other controls of the urine and faeces are also done.

d) Registration of doses

In most cases the worker will wear at least two types of dosimeter. The first type is the legal one and will register the official dose received by the worker. It can be of two types: a film- or a TLD-dosimeter. In film dosimeter, the dose incurred is determined by means of the degree of photographic film blackening. TLD-dosimeter, comprises a material which absorbs energy from the incident radiation. This energy escapes in the form of light when the material is heated upon reading. The amount of light emitted is then a measure for the dose incurred. This technology is superior to film dosimetry. If there is a risk of irradiation at the level of the hands or fingers, supplementary dosimeters of this type can be worn at the wrest and at one finger.

The inconvenient of the above dosimeters is that they can't be directly read. It's then difficult to control what dose has been received during an operation. It's the reason why a directly readable dosimeter has been developed. Some years ago the most common dosimeter of this type was the pen-dosimeter, that was in fact a small ionisation chamber. In Western Europe, this type of dosimeter has been replaced by an electronic dosimeter. In the most cases, the measurement technique is based on a semi-conductor (**see Fig. 2.9**).

These types of dosimeter allow to control the received dose, but give also an alarm when the person comes inside a higher radiation field. It's one of the best ways to control the application of the ALARA-principle.

2.3. Non nuclear safety aspects

2.3.1. *General safety principles*

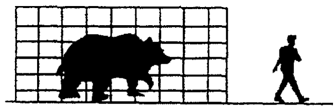
As clearly indicated in the regulations of a lot of countries concerning the safety in the workplaces the management of risks has to be performed in such a way that all kinds of risks are taken into account and that appropriate measures are enforced at different levels.

The nuclear sector doesn't "escape" to these statements. So, the general safety principles as shown here below have also to be applied.

risk

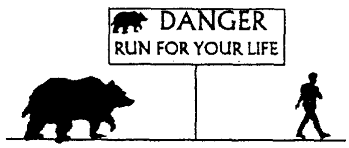
1. to eliminate risk

2. to reduce risk



3. collective protection

4. individual protection



5. to inform

These principles can also be expressed in terms of:

- prevention;
- protection.

We already have shown in § 2.1. how the nuclear safety can comply with the first goal and in § 2.2., we also indicated the means that are available to protect people against the ionizing radiations (exposition, contamination). Let us focus more specifically on the other hazards, the so called "non radiological hazards".

2.3.2. *The "non radiological hazards"*

The workers, depending upon the sector they are occupied in, are exposed to a lot of hazards such as fire, electrocution, uses of tools (crane, handling tolls, lift trucks,...).

In order to ensure their safety, one should apply the approach as described hereunder:

Safety assessment on the workplaces:

- identification and characterization of risks;
- to eliminate or to reduce the hazards;
- study of risks;
- evaluation of risks;
- risks analysis;
- definition of potential countermeasures;
- selection of the actions;
- enforcement of the actions;
- follow-up of these actions:
- analysis of the results;
- updating of the measures;
- incidents/accidents (feedback).

So, for each workplace, for each operation, we need:

- to perform a risk analysis;
- to decide for operational actions;
- to put these actions into practice;
- to assess the results;
- to bring corrections if needed.

Risk managers have furthermore to bear in mind the following remarks:

The risk can be defined as:

R = F x S

Where R = risk, F = frequency and S = security of the consequences.

Some methods have been developed to provide an estimation of these factors and to correlate the level of the resulting risk with the level of effort which has to be made (in terms of investment, priority,...) to comply with the principles described here above.

In some circumstances, safety managers have to be aware of the potential interaction between different sources of risks. For instance, as far as the nuclear sector is concerned, it has already been shown that some transfers from radiological up to non-radiological risks may occur. Furthermore, workers can also be faced simultaneously with two risks that are recognized as potentially severe. A good example can be found in reference, which describes the management of radiological risk and elimination of asbestos during the decommissioning of a nuclear power plant.

If the optimization principle is well known and almost applied by the "nuclear" safety manager, it should also be considered as a more global scale. In this sense, it's worthwhile to recall here that the risk management requires also the support of the management line. This imposes that the safety managers have sometimes to convince their hierarchy that means are needed to reach the safety objectives. For doing this, let us indicate that the literature on this subject provides some information that can be very useful if safety managers have to "justify" their approach. For example, if it's clear for everybody that the accidents distribution may look like a pyramid such as described in **fig. 2.10.**

This doesn't mean that the same level of effort has to be applied for each of the five "floors" of the pyramid. Indeed, some managers seem to limit their estimation to the direct costs of an accident (injury to the workers, repair of the installation) and they forget the so-called "indirect costs" (such as replacement of the worker, training, delay for the productions, brandimage consequences,...). These costs may be as high as 6 or 7 times more then the direct costs!

2.3.3. *Non Radiological Risks and Decommissioning*

It is obvious that a decommissioning operation implies several aspects of classic industrial safety. Such an operation can be compared to building construction yards, with additional constraints coming from the radiation and contamination protection aspects (like e.g. the wearing of gloves, overshoes, overall, full-face mask, pressurised suit, shielding, etc..). The classic risks of falling, load handling accidents, cutting tools wound etc. are obviously present in a decommissioning yard. Nevertheless, the decommissioning of a nuclear facility presents also some specific safety aspects which must also be taken into account.

The changing environment is the first one, as protective equipments can be dismantled, infrastructure equipments can be removed and sometimes structural integrity must also be regarded. Moreover, the operators working on a dismantling yard are not always fully aware of the new risks and dangers which can be encountered; this is mostly the case when former nuclear operators of the dismantled plant are mobilized for this new type of activity.

A second important aspect is the potential presence of toxic or dangerous materials present in the dismantled plant; like e.g. asbestos in thermal insulation or in cement, acids or chemicals remaining in tanks and piping or used as decontaminating agent. It is often needed to train the operators for being able to face these risks and to know which protection has to be taken.

The combination of industrial and radiological risks can also be a specific aspect of the operation, leading to adapted procedures and methods and to look for the way of minimum risk instead of focusing only on the radiological aspect of the safety.

2.3.4. *The influence of one safety aspect on the other one, and the ASARA approach*

Sometimes, the influence of one safety aspect on the other can be important, and one should then apply a broader concept than the ALARA principle to improve the safety of the operators and the environment. This will be highlighted by two typical examples from the BR3 dismantling project.

The first one is a quite typical case, where the radiological aspect was only considered by the operator. It was given the name of the "ladder syndrome" by our internal health physics department. Some operator who had to work in a controlled area with a ladder was wondering about the potential contamination of his tool. His ladder was made out of aluminium, so the decontamination should easily be performed afterwards. Then he considered that the rubber pieces placed at the extremities of his ladder should give rise to problems for further decontamination. These safety-related rubber pieces were placed to avoid the ladder to glide during use. Our worker decided to put some little plastic bags around the rubber extremities to protect them against the potential contamination. Then he went into the controlled area and used his tool, that glided! The worker fell and one of his legs was broken. He had to stay home for three months! This is a very simple example of interaction between radiological and non-radiological hazards. There are many other cases. The question is then "By optimisation of the radiological side of our works, do we not transfer some risks to the non radiological field?"

The second one concerns the removal of asbestos in places difficult to access. Elimination of the thermal insulation from the legs of the reactor pressure vessel at BR3. This set of operations took place during the month of February 1999. Workers belonging to the staff of BR3 performed the related tasks. This was the result of preliminary discussions with the administration of the Technical Inspection on the Workplaces. Indeed, due to the limited quantities of asbestos which had to be taken away and to the very well defined tasks to be performed, she concludes that the nuclear know-how and the safety features usually applied were adequate and that the managers of the BR3 didn't have to work with a licensed external firm. The estimated collective dose was 3,3 man-mSv and the received collective dose was 2,8 man-mSv (9 days and 10 workers).

Another asbestos removal yard was performed for the insulation of the primary loop. The work was performed as foreseen by a specialised and licensed company but some modifications were brought and some requirements were added too. For example, instead of working "top-down" as far as the removal of asbestos was concerned, the contractor accepted to begin in the middle of the steam generator. This decision was taken as a result of the pre-job ALARA study performed with the software VISIPLAN. Another example is related to the use of the full-face masks. After a few days, due to our additional check-up for internal contamination, it seemed that the masks in use in nuclear areas were more efficient and that they were more guaranteed for the safety of the workers. A third example copes with "at random" check-up in the Whole Body Counter for potential internal contamination. This has led to the evidence of malfunction (or misuses) of the personal protection equipment and brought to more severe requirements. On the other side, this supplementary measurement was very well accepted by the workers. Their fear, as far as nuclear risks were concerned, has really decreased! Finally, the dose for each worker was daily recorded and transmitted to the partners (contractor, BR3, Health Physics and Safety Department). Finally the operation took 35 days (instead of 50 as estimated) and the cumulative dose uptake amounted to 19.2 man-mSv instead of 89 as estimated.

These aspects of mutual influence of non-radiologcal and radiological safety, as well as new trends indicate the need for an extension of the ALARA approach to a broader extend, i.e. introducing the concept of ASARA principle, for As Safe As Reasonably Achievable.

2.4 References

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Chapter 2: Nuclear and non-nuclear safety aspects

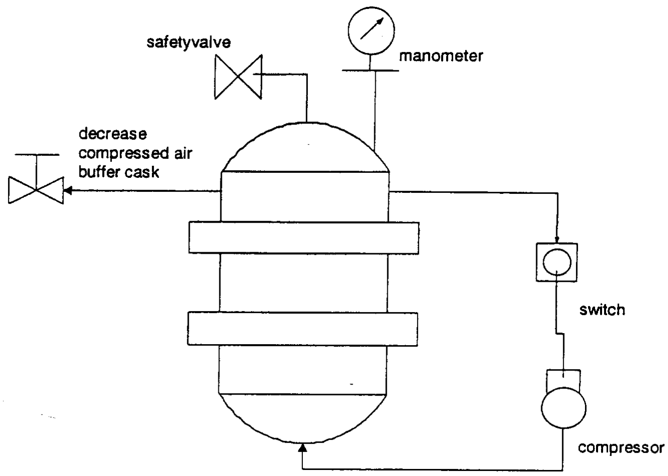

Fig. 2.1 Example Safety assessment

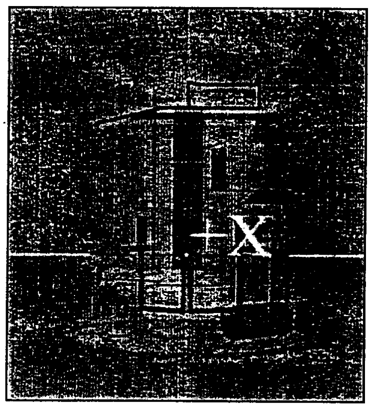

Fig. 2.2: Schematic representation of the containment building

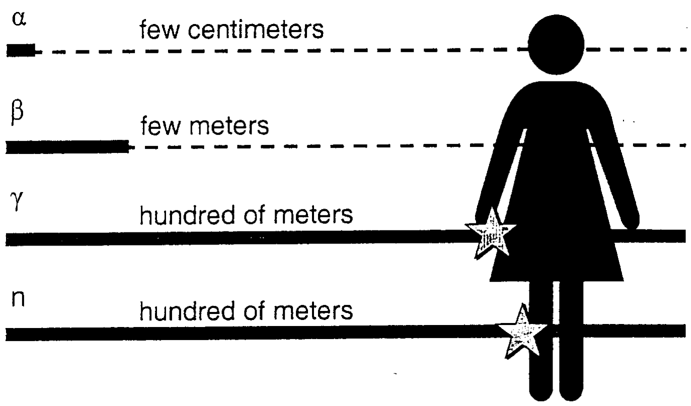

Fig. 2.3: Characteristics of the different types of radiation in the air

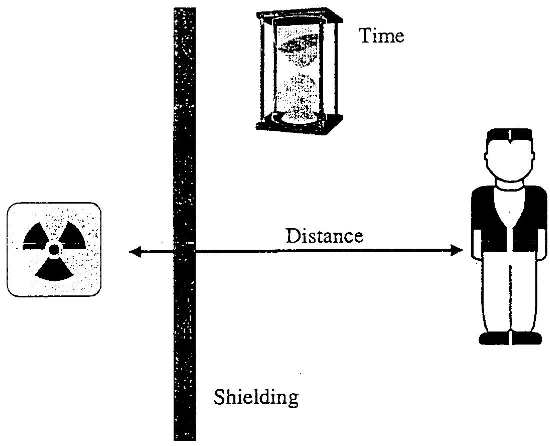

Fig. 2.4: Protections against irradiation

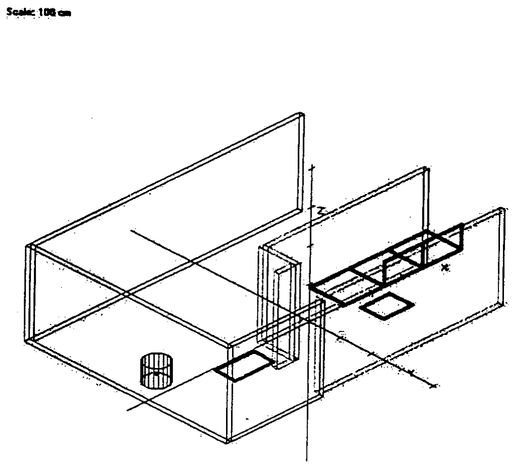

Fig. 2.5: Model Hot Cell

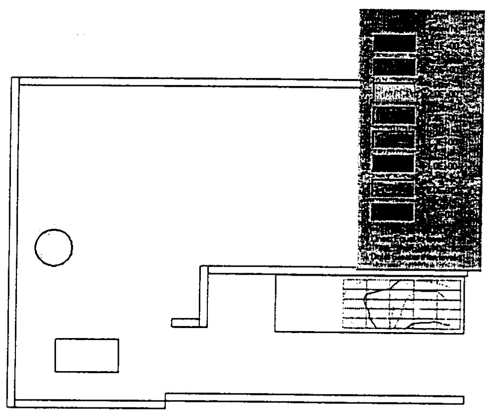

Fig. 2.6.: Calculation of a doserate field above the working table

Fig. 2.7: Overpressure packs

Fig. 2.8: Overall + Mask with P3-filter

Fig. 2.9.: Example of electronic dosimeter

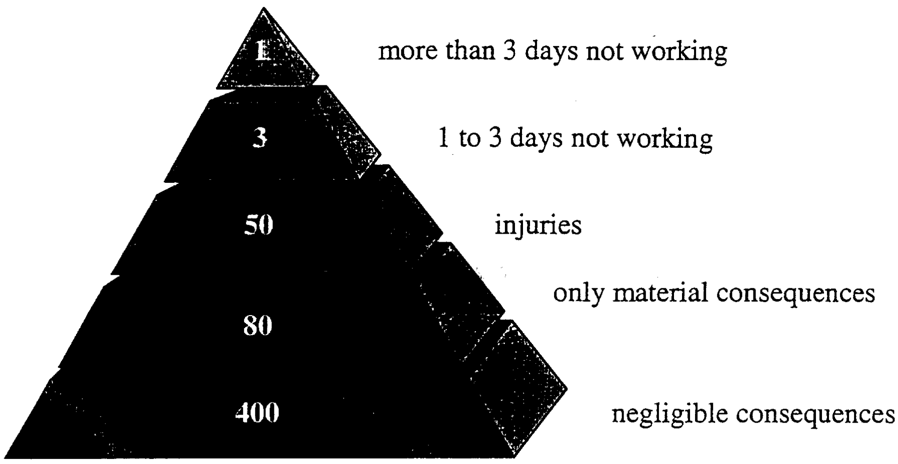

Fig. 2.10: Consequences of accidents/incidents

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Chapter 3

Regulatory Aspects of decommissioning

Chapter summary

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Regulatory aspects of decommissioning

3.1 Introduction

Decommissioning is a quite complex process, covering a several year period and a wide typology of activities. Main areas of activities are:
- safe operation and maintenance of nuclear installation under dismantling
- dismantling, such as cutting, decontamination, handling of removed components
- waste and material management, including storage and transport of radioactive material
- disposal of both radioactive wastes and non radioactive hazardous material.

Each of these areas requires suitable regulation to ensure that all tasks carried out during decommissioning are performed safely and with minimal impact on the workers, public and environment. In order to facilitate progress towards better understanding of the rationale behind and practical implication of decommissioning regulation, dialogue between regulators and operators has already begun. In this context, a specific workshop on these topics was held in Rome, 1999, jointly sponsored by IAEA, EC and NEA. Main purpose was to identify issues were international consensus is not yet reached and further discussions are needed.

3.1.1 Safety

Within their legislative framework, all States with nuclear programmes have some regulations for closure and subsequent decommissioning of nuclear facilities. In respect of regulation for safety, decommissioning work is not different from the design and operation of nuclear facilities and therefore the scope of regulation is often considered to be the same as during the operational period. In general, the same or very similar framework used for licensing, commissioning and operating nuclear facilities is also applied to the shutdown and decommissioning and provides a continued but flexible safety regime until the decommissioned site is released for new purposes.

Nevertheless, while the general criteria are surely the same, specific aspects need to be deeply examined and regulated. Therefore, considering the increasing number of ongoing decommissioning projects, in recent years both national and European authorities and administrations are developing specific rules for decommissioning.

Individual countries regulations are influenced variously by such matters as national policy on the future use of nuclear power, the continued availability of trained staff, societal issues associated with the effects of facility shutdown and D&D on neighbouring communities, and by the broader financial issues of how best to use available funds and when to deploy them.
On the European side there is a need for harmonisation of regulatory aspects, specially when implementation of activities can involve transboundary considerations (transport of wastes or of released material for example).

3.1.2 Environmental protection

While there has been a general assumption that compliance with radiological assessment requirements would ensure adequate protection of the environment, in recent years in European Union environmental and socio-economic aspects are formally included in the

Environmental Impact Assessment (EIA) process, as part of a nuclear power station decommissioning process.
The original EIA Directive (85/337/EEC) has been amended (European Council Directive 97/11/EC) so that an EIA is required for all new dismantling and decommissioning operations of nuclear power plants. The main principles involved in EIA under EU Directive are that it should take place for all private and public projects likely to have significant effects on the environment. The EU directive specifies minimum requirements for the information that should be included in the report and advice is given on how to make the decisions as to whether an EIA should take place.

3.1.3 Responsibility and functions of regulatory bodies

The risks involved in the decommissioning of nuclear facilities and the management of their wastes are:

- Technical (public, workers, equipment...)
- Societal (environment, future liabilities, welfare...)
- Financial (provisions, access to the funds, management of the funds.

The responsible groups involved in the policy and the regulatory framework for decommissioning are:

- the licensing authorities
- the government/states/region/local authorities
- the owner of the plant
- the contractors.

It is considered as important that responsibilities and participants be clearly identified by law in each country, possibly imposing a uniform guideline in the European context.
In general, the setting of national D&D policies and the establishment of legislation and regulatory requirements are carried out at national level by Government Departments or Ministries. Typically, these include Ministries for Trade and Industry, for the Environment, for Health, and for the Economy. The systems for developing legislation and regulatory arrangements may vary in detail depending upon constitutional arrangements of the countries and some countries make provision for the involvement of specific stakeholder groups. Nevertheless, it is generally true that in matters concerning nuclear power, the primary body for these issues is central Government.
In some countries, the ultimate regulatory authority remains with relevant Government Departments, who are advised by inspectorates charged to carry out site inspection, review of licences, monitoring, etc. In other countries, regulatory bodies are charged with enforcing the laws on nuclear safety, radiation protection and environmental protection independently of Government Departments, and they report directly to Government. In most countries, however, the regulatory bodies operate independently within a well-defined institutional mandate but answer to one or more relevant Government Departments, or to their equivalents in Federal States. Detailed arrangements vary widely across countries but they are usually some combination of the above examples.

Recently, an expert group at European level pointed out:

- In the case of transfer of responsibilities, e.g. to future generations, the Member States must verify that this transfer is feasible and that not only the responsibilities are transferred, but also means to achieve them: technical knowledge, financial provisions....

- Each member state must ensure that the decommissioning of each of its nuclear facilities be completed to the final stage determined by their Authorities.
- There is a particular responsibility for waste management: the member states must determine clearly who is responsible for waste repositories, and that all these responsibilities are fully covered in the longer term.
- Responsibilities associated with conventional risk shall also be considered (see **annex 3.1**).

In addition, a document issued from OECD/NEA raised the question whether current arrangements will be adequate for ensuring safety throughout the transition from operational to non-operational status, and over the long time scales associated with deferred decommissioning. Phased release of some parts of a nuclear facility could imply new regulatory issues.
Another item that has been arisen is the relationship between the environmental impact assessments and the safety case.
These, and others as the harmonisation of practices, are features to take into account to consider whether current arrangements will continue to be adequate for the long term.

The regulatory framework for the decommissioning process can be shortly outlined with the following levels:

- International Treaties and Conventions and International legislation, such as European Directives
- International Rules and standards (ICRP, IAEA, ...)
- National (and/or regional) legislations
- National standards.

The scope of the present Course is to outline the first and the second level, addressing specific attention to the main items that national legislation and standards have to deal with, including licensing procedures and funds raising mechanism. As it would be impossible, and even useless, to consider in detail each national legislation, only a few examples will be proposed (considering some specific aspects, the legislative framework for decommissioning in Russia is described in **Appendix 1**). This exercise will be enough to enable trainees to become familiar with main regulatory issues.

3.2 Safety and radiation protection criteria

In the field of decommissioning, as in all nuclear related activities, the main issue is the short and long term safety of the public and of the environment, and continuing to protect the health and safety of workers in the process. This is implemented by:

- Fixing dose limits for worker and people. These imply to fix related quantities as the contamination levels for air, water and soil and the exposure rates due to ionising radiation inside the working areas and the surround environment. Depending on the environmental characteristics of the site and on the public habits, these limits lead to define the discharge limits in the environment and limits for free release.
- Compliance with the ALARA principle. This means that even if the incurred individual doses are below national, legal dose limit, they should also be kept As Low As Reasonably Achievable, economic and social factors being taken into account.

- Waste classification and waste management. As consequence of the aforementioned statement, good waste management practices are essential for the workers safety and to limit the risk for the environment. This implies different procedures for different risk class of waste. Processes of treatment and conditioning have to be approved.
- National/Regional Storage/Repository safety requirements. Also this topic is a consequence of the first statement. Licence is needed for operating repository. Special licences in this context are needed for the waste containers, depending on their content classification.

3.2.1 International Conventions

Four important international conventions enforcing safety obligations have become operative (mainly under the auspices of IAEA), and they apply of course also to decommissioning activities. They are:

- The Convention on early notification of a Nuclear Accident
- The Convention on Assistance in the case of a Nuclear Accident or radiological Emergency
- The Convention on nuclear safety
- The Joint Convention on the safety of Spent Fuel Management and on the safety of Radioactive Wastes. The Joint Convention entered into force on 18 June 2001 and as of December 2001 had been ratified by 27 IAEA Member States. The Joint Convention contains a number of articles dealing with planning for, financing, staffing and record keeping for decommissioning. The Joint Convention requires Contracting Parties to apply the same operational radiation protection criteria, discharge limits and criteria for controlling unplanned releases during decommissioning that are applied during operations.

3.2.2 European Union

Countries in the European Union are also bound by the terms of the EURATOM Treaty. The Treaty establishing the European Atomic Energy Community (or EURATOM Treaty) is one of the founding treaties of the European Union. The Treaty was originally drafted in the 1950s and addresses the issues in the field of nuclear power that were relevant at that time. These include radiological protection of the work force and the public (Chapter III, Health and safety), the supply of uranium for the developing nuclear power sector (Chapter VI), the safeguarding of this fissile material to prevent it from being used for unauthorised military purposes (Chapter VII) and general aspects such as research and dissemination of information. However, the EURATOM Treaty makes little or no specific mention of aspects such as operational safety of nuclear power plants and radioactive waste storage or disposal facilities. This is probably because at the time the Treaty was drawn up, these were not major concerns.

Under the provisions of the EURATOM Treaty, the European Commission acquired the status of a supranational regulatory authority.

Article 30 of the Euratom Treaty entrusts the Commission with the task of devising "basic standards for the protection of the health of the workers and the general public against the dangers arising from ionising radiation" as:

(a). maximum permissible doses compatible with adequate safety;
(b). maximum permissible levels of exposure and contamination;
(c). the fundamental principles governing the health surveillance of workers.

Article 31 establishes the procedure for the Commission to work out the basic standards.
Article 32 establishes how the basic standards may be revised or supplemented.
Article 37 states that "Each Member State shall provide the Commission with such general data relating to any plan for the disposal of radioactive waste in whatever form as will make it possible to determine whether the implementation of such plan is liable to result in the radioactive contamination of the water, soil or airspace of another Member State. The Commission shall deliver its opinion within six months, after consulting the group of experts referred to in Article 31".
On December 6th 1999, the European Commission published a recommendation (1999/829/Euratom) on the application of this article. Section 1, subsection 9, of this Recommendation defines "any planned disposal or accidental release of radioactive substance associated with the dismantling of nuclear reactors and reprocessing plants" as falling within the Article 37 definition of the disposal of radioactive waste.

EU legislation can be passed as Regulations, Directives, Decisions and Recommendations that are implemented in different Member States by way of national legislation and regulation:

- A regulation has general application. It is binding in its entirety and directly applicable in all Member States.
- A directive is binding, as to the result to be achieved, upon each Member State to which it is addressed, but leaves to the national authorities the choice of form and methods.
- A decision is binding in its entirety upon those to whom it is addressed.
- Recommendations and opinions are not binding.

European Commission initiatives in the field of decommissioning nuclear installations are based on the above mentioned chapter 3 (article 37 et al) of the "EURATOM Treaty" and on the Directive of the Council 96/29/Euratom of 13 May 1996 laying down basic safety standards for the protection of the health of workers and general public against the danger arising from ionising radiation. This directive in its article 5 deals with "Authorisations and clearance for disposal, recycling and reuse of materials containing radioactive substances". A communication from the Commission (98/C 133/3 of 30.4.98) was issued in order to assist the Member states in transposing the Directive in national laws.
Other relevant EU documents are listed in References.

3.2.3 IAEA

Decommissioning and Decontamination has been included in IAEA's programme since 1973. In order to assist Member States in the implementation of adequate levels of safety in the nuclear practices, IAEA, under the terms of Article III of the Agency's Statute, which authorizes the Agency to establish standards of safety for protection against ionizing radiation, issues publications of a regulatory nature. These cover nuclear safety, radiation protection, radioactive waste management, the transport of radioactive materials, the safety of nuclear fuel cycle facilities and quality assurance. Publications are classified into two types: Safety Standard Series and Technical documents.

Safety Standards Series publications are categorized into:
- Safety Fundamentals, stating basic objectives, concepts and principles of safety and protection;
- Safety Requirements, establishing the requirements that must be fulfilled to ensure safety for particular activities or applications;
- Safety Guides, recommending actions, conditions or procedures for complying with these safety requirements.

Technical documents are categorized into:
- Technical reports, which deal with the consolidated experience and lesson learned, and are based on international consensus.
- Technical documents (TECDOC), which give preliminary information or deal with problems where international consensus is not yet reached. They do not provide specific guidance. They can be useful as reference.

In relation to decommissioning activities most significant standards are:
- the "International Basic Safety Standards for protection against ionizing radiation and for the safety of Radiation Sources" (Safety Series n 115, 1996); this standard was also sponsored by FAO, ILO, NEA-OECD, PAHO and WHO.
- Safety Series 111-F, 1995, "The Principle of Radioactive Wastes Management"; this document should serve as a guide in the development of radioactive wastes management national programmes, and deals from waste minimization items to disposal.
- Safety Series 52, "Factors relevant to Decommissioning of Land Based Nuclear Reactors Plants" (1980), sets out the now nearly universally adopted stages of decommissioning with the principal objectives of defining responsibilities and safety criteria, such as planning activities, radiation protection and quality assurance.

IAEA has recently issued Safety Requirements document and three Safety Guides applicable to decommissioning of facilities. The Safety Requirements document, WS-R-2, Predisposal Management of Radioactive Waste, including Decommissioning, contains requirements applicable to regulatory control, planning and funding, management of radioactive waste, quality assurance, and environmental and safety assessment of the decommissioning process.
The three Safety Guides are WS-G-2.1, Decommissioning of Nuclear Power Plants and Research Reactors, WS-G-2.2, Decommissioning of Medical, Industrial and Research Facilities, an WS-G-2.4, Decommissioning of Nuclear Fuel Cycle Facilities. They contain guidance on how to meet the requirements of WS-R-2 applicable to decommissioning of specific types of facilities. These Standards contain only general requirements and guidance relative to safety assessment and do not contain details regarding the content of the safety case. More detailed guidance will be published in future Safety Reports currently in preparation within the Waste Safety Section of the IAEA.

According to the above mentioned document categorization, a review process is ongoing in order to revise, if necessary, and re-issue previously issued safety standards. **Annex 3.2** gives the list of existing safety standards and their scheduled review process.
Annex 3.3 provides a list of technical documents specifically related to decommissioning.

3.2.4 ICRP

The International Commission on Radiological Protection, ICRP, is an independent Organization, established to advance for the public benefit the science of radiological protection, in particular by providing recommendations and guidance on all aspects of protection against ionising radiation.
ICRP offers its recommendations to regulatory and advisory agencies and provides advice intended to be of help to management and professional staff with responsibilities for radiological protection. While ICRP has no formal power to impose its proposals on anyone, in fact legislation in most countries adheres closely to ICRP recommendations.

ICRP publications concerning specifically decommissioning activities are Publication 77 "Radiological protection policy for the disposal of Radioactive Waste" and Publication 81 "Radiation protection recommendations as Applied to the Disposal of Long-lived Solid Radioactive Waste". ICRP Publication 60, 1990, which contain definitions of some new basic quantities, does not particularly deal with radioactive waste problems. However, in the general system of radiological protection, the optimization of protection, as well as dose limits, now include the concept of potential exposure. It is expressed as the likelihood of incurring exposures where these are not certain to be received. They should be kept as low as reasonably achievable (ALARA). The ICRP Publication 60 has been taken into account in developing the European Council Directive 96/29/Euratom.

3.3 Material management and release

3.3.1 General issues

The management of materials and the related radioactive waste during D&D of nuclear facilities are key considerations in the planning and timing of the D&D activities. The availability of well proven procedures for each phase of the process, the set up of clearance levels and the availability of waste disposal facilities are important issues in considering how and when starting the facility dismantling so generating large quantities of wastes.
Fig. 3.1 shows the typical path for material management.

The International Basic Safety Standards (BSS) [IAEA, 1996] lay down some concepts that are important for a full understanding of main problematic related to material management.
Human activities that add radiation exposure to that which people normally incur due to background radiation, or that increase the likelihood of their incurring exposure, are termed "practices" in the BSS. Human activities that seek to reduce the existing radiation exposure, or the existing likelihood of incurring exposure, which is not part of a controlled practice, are termed "interventions". The BSS provide the basis for a regulatory system for the control of radiation: one part of the system applies to practices, another to interventions.
Moreover, it is essential to well understand following concepts that have been introduced in the more recent regulatory documents, to which **Fig. 3.2** refers:

- **Exclusion:** covers activity sources not amenable to control, and refers to "any exposure whose magnitude or likelihood is essentially unamenable to control through the requirements of the Standards". The characteristic of an excluded exposure is not whether or not it is of "concern", but rather that the exposure is simply unmanageable and uncontrollable through regulations. Examples are K-40 in the human body, cosmic radiation, etc.; a more specific example is the gaseous discharge, through a building ventilation system, of radon and associated daughters arising from the ground or construction materials.

- **Exemption:** this term had earlier been used to denote all radioactive material placed outside regulatory control because of the low risk they give rise to. Now the conditions for exemption have been better precised, and there is an international consensus above:

  - the individual radiation risk attributable to the exempted source should be sufficiently low as to be of no regulatory concern
  - the collective radiological impact resulting from the exemption should be sufficiently low as not to warrant regulatory action
  - the exempted situation should be inherently safe, i.e., with no appreciable likelihood of scenarios that could lead to a failure to meet the above criteria.

  The ICRP has recommended the use of "intervention exemption levels", in order to avoid unnecessary restrictions: this could apply to decommissioning activities.

- **Clearance** has been used, after the restriction of the meaning of "exemption" mentioned above, to denote material that has been released from regulatory control. Sources can be released from regulatory requirements provided that it can be demonstrated that they present trivial risks to individuals and populations.

  The clearance concept is different from the exemption concept, since the materials subject to clearance are already under regulatory control until the regulatory authority clears them. It is responsibility of the Regulatory Authority to establish requirements for clearance and verify compliance with the requirements.

  The concept of clearance implies a removal of restrictions so that the cleared material can be treated without any consideration of their radiological properties. However, there is also the possibility of clearing material under specified conditions. In this case one can speak of "conditional clearance", where the applications of conditions ensures adequate degree of radiological protection to the public. An example of conditional clearance is reuse or recycling in the nuclear industry.

It is difficult to relate the dose received by individuals to a specific practice, or to the levels of radioactivity involved in a practice. This applies even more in the case of clearance than in the case of a fully regulated practice, since the clearance criteria must be defined for a largely hypothetical environment
For this reason, clearance are usually not expressed in terms of individual or collective dose, since it is not practical to measure these parameters at the operational level. Rather, they are expressed in terms of derived quantities (activity concentration and/or total activity), that are directly measurable so that compliance with the requirements can be verified. The proposed clearance levels in terms of derived radioactivity levels have to take into account the most critical scenario.

The clearance requirements may relate to the total amount of material, to the total activity, to the radio nuclide specific total activity and concentrations, or to the physical and chemical form of the material subject to clearance.

In the context of material management, record keeping is a fundamental issue. Just to give some examples, records have to be foreseen for the health physics service (records of authorised discharge, records of personnel dose data, records of radiation monitor, etc.), for the personnel administration (records of work permit), for the contractors management etc. In the material management area, the record keeping relates to the whole of the software/hardware and procedures which guarantee the correct management of materials and wastes, in terms of safety, final waste minimisation and legal surveillance. This implies the registration of the material (or component) characteristics that are the results of the characterisation (location, weight, volume, contamination, etc.) the process/treatment made, the package or container used for transportation inside the plant, finally the final route that can be free released, disposed for recycling or sent to the final repository. A major item of record keeping system is to assure the traceability of the materials.

3.3.2 International and European regulations

IAEA
In 1988, the IAEA and the Nuclear Energy Agency (NEA), in co-operation, issued Safety Series No. 89 to recommend a policy for exemptions from the basic safety system of notification, registration and licensing that form the basis of regulatory control. Safety Series No. 89 suggests:

- a maximum individual dose/practice of about 10 μSv/year,
- a maximum collective dose/practice of 1 manSievert/year

to determine whether the material can be cleared from regulatory control or other options should be examined.

A methodology to apply these principles on the recycling or reuse of material from nuclear facilities was subsequently presented. [IAEA Safety Series No. 111-P1.1 "Application of Exemption Principles to the Recycle and Reuse of Materials from Nuclear Facilities" (1992) and Safety Series 111-G-1.5: "Clearance Levels for Radionuclides in Solid Materials (1995)].

The first document presents exemption/clearance limits differentiated for typical materials, such as concrete, steel, aluminium etc. The second document distinguishes, for the first time, terms of exemption and clearance and introduces the concept of conditional clearance.

Both documents were part of the input in the IAEA process of establishing unconditional release levels for solid materials, laid down in the interim TECDOC-855, 1996; this document, while still characterized as an interim report, is considered as a reference in the regulatory framework mainly outside the EU.

In 1996, the above mentioned standard safety series n 115 (a revised International Basic Safety Standards) gives a list of nuclide specific exemption values (both quantities and concentrations).
The issue of clearance level is being addressed in a new Safety Guide being prepared by the IAEA dealing with the Scope of Regulatory Control. This Safety Guide will attempt to rationalise levels of radioactivity subject to exclusion, exemption, discharge, recycle, contained in commodities, and released from regulatory control, taking into consideration levels of radioactivity in naturally occurring radioactive materials. This Safety Guide is scheduled to be completed late in 2002 or in 2003. Obviously, because much material arising during the decommissioning of nuclear facilities may be only slightly contaminated with radioactivity, this is really an important matter for decommissioning

EC

The mentioned EC Council Directive of May 1996, lays down its BSS for radiation protection, with nuclide specific exemption values very similar to those in the International BSS. However, the EC BSS makes a difference between "practices" covering processes utilising the radioactive, fissile or fertile properties of natural or artificial radionuclides (i.e. the nuclear industry) and "work activities" where radioactivity is incidental, but can lead to significant exposure of workers or the public. As already mentioned, another important reference for EC regulation is ICRP 60 that discusses the concept of exemption from regulatory control in Radiological Protection Criteria.

Community guidance has been produced for the application of the clearance concept for the reuse, recycling or disposal of materials arising from the dismantling of nuclear installations (metals, buildings and building rubble). Clearance for disposal has been looked into only for building rubble. Landfill disposal in general is considered to be a matter of national competence rather than an issue for the Community (even though transboundary movements of waste may need to be taken into consideration).

The clearance of metal scrap on the other hand has definitely a transboundary impact and harmonisation of the clearance levels would be highly desirable. This can be achieved within the current Directive only by voluntary co-operation between Member States. The Commission can take further regulatory initiatives, e.g. propose specific exemption values for the placing on the market of metal scrap. Monitoring at the borders of the EU may be encouraged. Minimum detectable dose rates for this purpose can be derived from the specific exemption values.

Member States may also find it useful to define general clearance levels for any possible application. Such a lower boundary to materials under regulatory control would again usefully be established at Community level.

In 1999, a report with opinions of experts was published by EC (EUR 18860). With reference to the clearance issues, expressed opinions were:

- "concepts of exemption, conditional and unconditional clearance should be maintained.....Although a disparity may be perceived between the values, they need to be maintained if we want to address the number of issues that decommissioning activities raise. The number of various concepts or criteria create confusion; therefore, one needs to find the correct language to put them across
- industrial concrete is more and more recycled in the construction industry instead of disposed ......On going work on release criteria for contaminated concrete should be pursued
- the group experts raised the question of clearance levels and detection limits. The detectors, that ....are used by scrap dealers are of sufficient sensitivity to detect radiation below clearance levels. This issue should be considered when developing a strategy of information of the public...
- the group experts stressed the importance of cooperation and clarification of release principles and criteria with other international organisations"

Specific EC recommendations have been prepared for the clearance in different fields of application:

- Radiation Protection 89 "Recommended radiological protection criteria for the recycling of metals from the dismantling of nuclear installations" [1998]. The guidance makes proposal for recycling and reuse ferrous, copper and aluminium metal scrap and alloys of these metals .The EC approach provides two options for releasing material:

- direct release based only on surface contamination,
- melting at a commercial foundry followed by recycle and reuse. Mass specific and surface specific levels are provided.

- Radiation Protection 114 "Definition of clearance levels for the release of radioactively contaminated buildings and building rubble Final Report (1999). The criteria for derivation of clearance levels is related to the further purpose of usage or handling.
- Radiation Protection 113: "Recommended radiological protection criteria for the clearance of buildings and building rubble from the dismantling of nuclear installation" (2000). The document takes again the concept of clearance presented in RP 89 and in RP 114 and develops three sets of clearance for buildings and buildings rubble, due to the further purpose of usage or handling
  - clearance for buildings for any purpose (reuse or demolition)
  - clearance for buildings for demolition only
  - clearance for building rubble.
- Radiation Protection 122: "Practical use of the concepts of clearance and exemption" (2000). In this guidance the Commission introduces the notion of general clearance levels. The difference between exemption and clearance are clarified, the distinction between general clearance (unconditional) and specific clearance (conditional) are explained.

Table 3.1 provides clearance levels from different references and **Table 3.2** and **3.3** provide limits for release in some European countries: The only purpose of these tables is to show that wide differences exist, due to different approaches, definition of scenarios, and criteria. A great room for homogenisation still exists.

3.4 Waste classification and management, transport and disposal

Materials issued from the "Material Management Process" during decommissioning activities, as it has been shown, can either considered as "cleared" or "radioactive waste". From a regulatory point of view the waste management approach can be divided into three main areas: pre-disposal management, transport, disposal.

3.4.1 Waste classification and management

In the last decade, the waste classification and management issues have increased in importance, due mainly to the large amount of waste deriving from the current decommissioning projects world-wide. At present almost each country has its own strategies and policies about waste classification and waste management criteria and an harmonisation is expected. The strong demand of homogeneous waste classification and reporting is emphasised by the need of international comparisons and studies on the topic of radioactive waste.

As for reporting, IAEA, in order to provide support to the international harmonisation on this matter, issued the TECDOC-1097 "Maintenance of Records for Radioactive Waste Disposal" (1999) providing guideline for the implementation of national system for record keeping and record management.

Classifications are based on both qualitative and quantitative criteria, in which wastes are grouped according to their origin, activity content and life, radiotoxicity, thermal power. But, notwithstanding waste classification varies from one country to another, in general terms it is possible to say that nuclear power plants generate the following types of waste:

1. High level waste (HLW): such waste requires substantial shielding and in terms of volumes or weight it represents the smallest amount of waste (lower than 1%) issued from decommissioning. This may consist, for example, of residues generated from reprocessing of spent – or used – nuclear fuel. A special disposal site is needed for this type of waste. The waste must remain isolated for thousands of years.

2. Intermediate level waste (ILW): which represents some 10% of the total generated radioactive waste and requires shielding with concrete and/or steel. ILW typically comprises of outer components stripped from spent fuel elements, the fuel canning and filters and resins used in the clean up of gaseous liquid waste.

3. Low level waste (LLW): which in general accounts for almost 90% of all radioactive waste produced in the decommissioning of a NPP. LLW can be handled with little risk and without radiation shielding. Low-level waste is less hazardous than HLW and ILW, and can be shipped to disposal facilities (where in some cases it is packaged, in some others conditioned) then buried in trenches, and covered with soil.

A special issue concerns the Very Low Level Waste (VLLW), that carries low potential hazard. Generally, the bigger quantity of wastes coming from decommissioning has low potential hazard. Nevertheless there is presently no internationally agreed definition of VLLW and therefore in countries where they are not regulated they have to be treated at the same level of Low level waste. In countries where a specific definition and regulation exist, they can be disposed in separate repository, with less requirements, or in some cases they could be in exemption condition.

Table 3-4 illustrates the classification scheme proposed by the IAEA in Section 3 of IAEA Safety Guide 111-G-1.1(1994), "Classification of Radioactive Waste". The proposed classification scheme is based only on quantitative criteria, in which wastes are grouped according to the safety aspects of their management, especially disposal options.

Also on this basis, on 15 September 1999, the European Commission adopted Recommendation 1999/669 on a classification system for solid radioactive waste (Official Journal of 13 October 1999). This Recommendation aims to harmonise methods of classification of such radioactive waste between Member States in order to facilitate their co-operation within the common market and the free movement of goods and services. This system should be used to provide information to the public, the national and international institutions and the non-governmental organisations on solid radioactive waste.
The Member States and their nuclear industry are invited to adopt a common classification system of radioactive waste in order to improve national and international communication as well as to facilitate information management in this field.
The European Commission proposes a classification system for radioactive waste management based on the characteristics and properties of the waste involved, as well as their potential effects on the public and the environment. Three principal categories have been foreseen; the proposed classification is summarised as follows:

- **Transition radioactive waste**

Type of radioactive waste (mainly from medical origin) which will decay within the period of temporary storage and may then be suitable for management outside of the regulatory control system subject to compliance with clearance levels.

- **Low and Intermediate level waste (LILW)**

In LILW the concentration of radionuclides is such that generation of thermal power during its disposal is sufficiently low. These acceptable thermal power values are site-specific following safety assessments.

  - Short-lived waste (LILW-SL)

This category includes radioactive waste with nuclides half-life less than or equal to those of Cs-137 and Sr-90 (around 30 years) with a restricted alpha long-lived radionuclide concentration (limitation of long-lived alpha emitting radio-nuclides to 4 000 Bq/g in individual waste packages and to an overall average of 400 Bq/g in the total waste volume).

  - Long-lived waste (LILW-LL)

Long-lived radionuclides and alpha emitters whose concentration exceeds the limits for short-lived waste.

- **High level waste**

Waste with such a concentration of radionuclides that generation of thermal power shall be considered during its storage and disposal (The thermal power generation level is site-specific and this waste is mainly forthcoming from treatment/conditioning of spent nuclear fuel).

Even if the Recommendation provided that national systems of waste classification might be used in parallel with the community system until 1 January 2002, no efforts in this way have been carried out by all countries.

Special consideration is to be devoted to spent fuel, still containing fissionable isotopes, which could be further used to produce energy. This is the key reason because spent fuel may not be considered as waste but it is a category on its own. Independently from the final option adopted, i.e. reprocessing or disposal, specific regulations exists for the interim storage, always needed in both options. Regulations are developed for the storage building and for the containers, namely casks.
References from [3.32] to [3.34] deal particularly with spent fuel. Reference [3.35], specific for the transport of radioactive material, applies also to the spent fuel.

Considering the general item of the wastes management, some interesting points that refer to the regulatory level are developed in the Communication from the UE Commission to the Council (COM 94) 66 that for the first time tried to define a Community Strategy for radioactive waste management:

- Proximity principle for radioactive wastes: this principle, that is at the base of the management of non radioactive wastes, has to be evaluated case by case for radioactive wastes; treatment, storage and disposal facilities will have to be centralized for economic, safety and environmental reasons. The optimization can be considered at a Community level.
- Self sufficiency in radioactive waste disposal: while self sufficiency at the level of the Community is evident, the document outlines how it seems regrettable, and at least premature, to deny the possibility of assistance to another country of the Community in specific cases, notably those putting at the stake nuclear safety. A regional approach, involving several countries, could offer advantages. The

development of a solidarity approach to disposal was therefore decided. (This approach was not prevented or hindered by the Joint Convention on the Safety of Spent fuel management and on the Safety of the radioactive wastes management).

- The radioactive wastes equivalence concept: some countries, which have specialized nuclear facilities not commonly available, are processing or conditioning wastes from other countries as a result of commercial arrangements. In such a situation one may consider, or indeed may be compelled, to return "equivalent wastes". Guiding principle for radioactive wastes equivalence should be set up, as presently they are not deeply dealt in European regulations.

3.4.2 Waste transport

The objective of the regulations is to protect people and the environment from the effects of radiation during the transport of radioactive material.
Protection is achieved by:

- containment of radioactive contents;
- control of external radiation levels;
- prevention of criticality; and
- prevention of damage caused by heat.

The fundamental principle applied to the transport of radioactive material is that the protection comes from the design of the package, regardless of how the material is transported.

Since 1961 the International Atomic Energy Agency (IAEA) has published advisory regulations for the safe transport of radioactive material. These regulations have come to be recognised throughout the world as the uniform basis for both national and international transport safety requirements in this area. Requirements based on the IAEA regulations have been adopted in about 60 countries, as well as by the International Civil Aviation Organisation (ICAO), the International Maritime Organisation (IMO), and regional transport organisations.
The latest IAEA's recommendations are contained in the document 'Regulations for the Safe Transport of Radioactive Material' TS-R-1(2000 - ST-1, Revised).
These recommendations are then incorporated by the various modal organisations into their own regulations: ICAO for air, RID for rail, ADR for road, IMO for sea transports, UPU for mail, etc (all EU's Member States - with the exception of Ireland - are parties to the ICAO, ADR RID and IMO agreements).
At this stage they become binding on the parties to the agreements but before coming into force they generally still have to be incorporated into national regulations, as national laws seldom make direct reference to international regulations, and moreover they can introduce additional requirements.

While recommendations on the transport of radioactive materials are drawn up by the International Atomic Energy Agency and transposed into the national legislation of each country, the European Community must ensure that these provisions are in conformity with the Council Directive on radiation protection and that they facilitate the functioning of the internal market. The legal basis for the actions of the European Commission in the field of transport of nuclear materials has its origin in the EC Treaty, specially Title V on the common transport policy, and in the EURATOM Treaty where Chapter III provides the legal framework for setting basic safety standards on radiation protection.

The last updating of this legislation is the already mentioned Council Directive 96/29/Euratom: Basic Safety Standards for the Protection of the Health of Workers and Public against the Dangers Arising from Ionising Radiations.

One should note, however, that the Basic Safety Standards set in Directive 96/29 are slightly different for some nuclides from those on which the IAEA's transport recommendations are based.

EU legislation specific to Transport of Radioactive Materials are listed and shortly commented in **annex 3.4.**

3.4.3 Disposal

Disposal is defined as the emplacement of wastes in an approved, specified facility without the intention of retrieval, but often with retrievability requirements. The objective of disposal is to provide sufficient isolation of wastes to protect humans and environment and not to impose any undue burden to future generations.

Responsibilities in disposal activities are defined along some basic principles such as:

- disposal of radwaste is a national responsibility and is usually done within the borders of the relevant country;
- the generation benefiting from nuclear power needs to implement a safe solution, avoiding to pass interim solutions to the next generations;
- the nuclear plant operators have this responsibility;
- the agency in charge of waste processing and disposal share also this responsibility;
- the operation licence of a NPP shall require to define and implement an adequate solution and a schedule for the disposal.

The tasks of the regulatory bodies are to ensure that operators and the agency in charge of waste processing and disposal live up to their responsibilities, to clarify requirements and to participate in the licensing process. They supervise and provide impetus to both plant operators and agency in charge of waste processing and disposal. Regulatory aspects concerning disposal are mainly related to radioprotection, as well for the design as for the operation of the repository, and to retrievability.

Because of different legal and regulatory structures and requirements, repository licensing process differs among countries: for instance, in Germany a single licensing process covers construction, operation and closure of a repository, whereas several licensing steps are required in other countries.

3.5 Quality Assurance

The QA requirements for the decommissioning aim at warranting that technical activities are carried out in a controlled way. As a consequence also for the decommissioning the reference codes are those of the ISO 9000 series.
Applying the ISO codes to decommissioning means that all activities having an impact on the health and safety of workers, the public or environment must be carried out in accordance to written procedures and with responsibilities well defined. Important subjects for the QA plan are those related to the waste packaging, to the related accounting and finally to the conservation of the data in the long time.

The Organisation and the principles of the procedures must be described in a Q.A. program. Usually the QA program for a decommissioning project will be adapted from that used during operation, putting more emphasis on those aspects associated with the management of wastes. The QA program should also describe how to manage an organisation, and required staff, that will evolve throughout the decommissioning program due the evolution of the status of the facility, thus allowing reduction in costs, staff and administration.
Most of the QA requirements related to specific aspects of decommissioning are presented in the same reports presenting the other technical requirements.
In **Table 3.5** most important documents where specific QA guidance can be found are listed.

3.6 Licensing

3.6.1 Licence for decommissioning

Decommissioning involves all the administrative and technical operations allowing to withdraw a facility from the list of licensed facilities. As for all practices or interventions, a specific licence is mandatory, as a formal, legally prescribed document issued to the applicant by the national regulatory body to perform specified activities.
The applicant shall provide documents to enable the national control authority and, in case, their authorised experts (advisors) to evaluate that the overall objectives of the project can be met in the respect of existing rules.
Generally, the decommissioning licence has not to be seen isolated: depending on the starting situation, the licence is a continuation of the operational licence (removal of fuel elements, further operation of the ventilation systems etc.). In many countries, this licence can be considered as an "amendment licence" in which the new plant conditions, including the necessary works and the required safety measures, are defined. In fact when a NPP is permanently shut down, some regulatory requirements (valid during its operation life) are currently applicable (as aforementioned, if the fuel has not been removed from the reactor vessel), but some others are no longer applicable, due to the fact that the nuclear power plant will enter a permanent shut down stage leading to the early decommissioning stage.

The licensing procedure is quite different from country to country, and is generally quite a complex process. While the final responsibility for the licensing generally is up to the Government, a good number of Governmental Bodies, Advisors, Local Authorities etc is involved. **Fig 3.3** shows possible relationships in the process.
The decommissioning licence usually specifies the final stage of decommissioning to be reached, the intermediate stages and the time frame. Such a specification conforms with the decommissioning policy adopted by the country.
The application for the decommissioning licence includes an environmental impact evaluation: environmental aspects have to be checked more and more in the process; following paragraph deals with this item specifically.
The national regulations often do not address emergency plan requirements. In fact, it is obvious that, once the nuclear fuel has been removed from the plant, the potential risk posed by this plant greatly decreases. The same is true for the radiological risk during dismantling. By adapting emergency plans to the different steps of decommissioning, not only would some savings be induced but some possible public concerns in the plant vicinity area could also be assuaged. Furthermore, the work assigned to manpower in charge of dismantling could be facilitated.

A big item in the licence process is linked to the different steps that are required for the delicensing from the operational status up to the release of the site. This should result in a compromise between the regulatory need for a monitoring of activities and the global efficiency and simplicity of the process.
Experience suggests that the regulatory process is essential and Licensing procedures are necessary and unavoidable. Furthermore in some countries, these may be subject to a difficult, technical and political debate. Consequently specially in countries where a decommissioning of NPP has to take place and the Regulatory process and Licensing procedures are not established yet the problem should be faced in advance in order to avoid delay and backlogs.
Annex 3.5 presents typical "golden rules" for a good process of licensing: these are derived from the EWN experience, but a good number of items is surely to consider as of international consensus.
Fig. 3.4, **3.5**, **3.6**, **3.7** show licensing procedures for Germany, Italy, France and USA.

3.7 Environmental protection

3.7.1 General EIA requirements and Guidelines

Environmental and socio economic aspects are formally included in the Environmental Impact Assessment (EIA) process, as a key part of the nuclear power station decommissioning in EU. Requirements for EIA within the EU are set out in Council Directive 97/11/EC of3 March 1997 amending Directive 85/337/EEC on the assessment of the effects of certain public and private projects on the environment. The two Directives have to be read together to find out the obligations of Member States.
The main principles involved in EIA under the amended Directive are that it should take place for private and public projects likely to have significant effects on the environment, and that development consent will be dependent on an adequate EIA having been undertaken.
Under paragraph 2 of Annex I of the Directive, EIA is always required (except in exceptional cases covered in Articles 1.4 and 2.3) for:
Nuclear power stations and other nuclear reactors including the dismantling or decommissioning of such power stations or reactors (except research installations for the production and conversion of fissionable and fertile materials, whose maximum power does not exceed 1 kilowatt continuous thermal load).

Although EIA will always be required for decommissioning of nuclear power reactors, EIA for decommissioning of other type of nuclear installations (dealt with in paragraph 3 of Annex I of the Directive) will be at the discretion of Member States. Special consideration is to devote to paragraph 3 of Annex II of the Directive, according to which Member States need to decide, based on a case-by-case examination or by setting thresholds or criteria, whether EIA will be required for Installations for the processing and storage of radioactive waste. Under paragraph 3 of Annex I of the Directive EIA is always required, among other, for the final disposal of irradiated nuclear fuel, solely for the final disposal of radioactive waste, solely for the storage (planned for more than ten years) of irradiated nuclear fuels or radioactive waste in a different site than the production site.

The Directive specifies minimum requirements for the information which should be contained in the report of an EIA and advice is given on how to make the decision as to whether an IA should take place, a decision process known as 'screening'.

The process for EIA is made up by the key stages outlined in **Fig. 3.8.**
After screening, the scoping stage establishes the content and the extent of the Environmental Information to be submitted to the competent Authority under the EIA Procedure. In some states the Competent Authority is required by legislation to give an opinion on the scope of the EIA, whether or not the developer requests this. In addition the scoping stage may be extended to include the specification of criteria for assessing the significance of any potential impacts that may be identified during the course of the EIA.
Following main stages are the Environmental Studies, their submission and review , and the final decision.
European Commission (DG Environment) has provided specific guidance on a number of aspects of the EIA process:

- EIA - Guidance on Screening (EC, June 2001)
- EIA - Guidance on Scoping (EC June 2001)
- EIA Review Check List (EC June 2001)
- Guidelines on the Assessment of Indirect and Cumulative Impacts as well as Impact interactions (EC 1999) (These guidelines make no mention of decommissioning, nor do they contain any reference to nuclear installations. However, the advice given on approaches to assess indirect impacts, cumulative impacts and impact interactions are of general relevance).

A specific study established under an EC Contract has been recently issued and it gives details on requirements and on the status of implementation of the EIA procedures and Directive in Member Countries as well in applicant Countries.
Fig 3.9 is obtained by this report; it shows main steps after the first screening phase. Phases from 1 to 10 deal with the scoping phase, that, in order to be effective, need to consider different possible alternatives and to involve all stakeholders. A final scoping report is then produced and made publicly available; about one year of work is expected.

Once the scope for the EIA has been defined, the developer may proceed with the environmental impact evaluation. In order to make an assessment of the potential environmental impacts and define ways to mitigate or eliminate them, as well as how to ensure their proper monitoring, a series of steps need to be undertaken:

- Determination of the environmental baseline: in order to predict the potential environmental impacts of the proposed project, the initial environmental conditions must be known. Environmental damage varies according to the initial conditions: The definition of the baseline conditions (e.g. meteorological conditions, geological formations, groundwater flow direction) also provides data necessary to make impact predictions.
- Impact identification: Potential environmental impacts are identified at this stage.
- Assessment of the significance of impacts: Significance can be gauged relative to environmental standards or to public perception for some more subjective impacts. In particular it must be made clear whether predictions represent best estimate or worst case scenarios, although in most cases it will be appropriate to present both.
- Development of mitigation measures.
- Listing of residual impacts: mitigation can rarely remove all the identified impacts and a clear statement has to be made about the impacts which would still occur even after the implementation of the mitigation measures.
- Development of a monitoring plan: In recognition of the fact that the evaluation of impacts is subject to uncertainties, it is appropriate to develop a monitoring plan which would be able to check that the residual impacts identified are the only ones which occur.

The product of all the environmental studies taking place in Steps 11 to 16 is reported in the EIS, which typically is written as the evaluation work is taking place. The EIS is the public report of the study that has taken place and needs to convey the information gathered in an understandable way.
After the EIS is produced, it may be issued as a draft version and subjected to a public review. This review is recommended when the project has proved to be controversial, but the stage may be omitted if there is already a large degree of agreement. After the final EIS is produced (Step 17), it must be made publicly available and submitted to the competent authority (Step 18).
The EIS review is normally the responsibility of the competent authority. Each country has defined the procedure for undertaking the EIS review; this normally involves the public (e.g. allowing them to submit comments, organising public meetings or hearings) and/or expert committees (e.g. the EIA Commission in the Netherlands). Sometimes, although the review is formally the responsibility of the competent authority, public meetings may need to be organised by the developer.

Depending on the country, the decision made by the competent environmental authority may or may not be binding on the sectoral authority (e.g. the nuclear safety authority), i.e. the decision on the EIS may be the sole decision required in order to authorise decommissioning or may be but another source of evidence feeding into the sectoral decision-making process. Decisions may approve the proposed development, reject it, or approve it subject to meeting certain conditions. In any case, the EIS decision should be justified and be made publicly available. Once the decision has been made the EIA process can be said to be complete although the monitoring programme must be implemented and the results of this programme must also be made publicly available.

3.7.2 Other regulatory items

A Common Position was adopted by the European Council on March 30th 2000 with a view to adopting a Directive on the assessment of the effects of certain plans and programmes on the environment.
Strategic Environmental Assessment (SEA) will be required under this Directive for plans or programmes (or their modifications) which are likely to have significant effects on the environment in a number of specific sectors including the energy sector. The Directive applies to plans or programmes 'required by legislative or administrative provisions' and 'which set the framework for future development consent of projects', e.g. development plans drawn up by local or national governments. The intent therefore is that the provisions of this Directive should be complementary to those of the EIA Directive which relates to the assessment of specific projects.

In addition to the requirements of the amended EIA Directive, some international conventions to which many EU countries are signatories set further public participation requirements. These include:
the UNECE *Convention on Environmental Impact Assessment in a Transboundary Context* (known as the Espoo Convention) which was signed on 25 February 1991 and came into force in 1997.
Article 7 of the EIA Directive is designed to implement the provisions for public participation prescribed by the Espoo Convention. A notification must be made to potentially affected Parties in the case of listed activities that are likely to cause an adverse transboundary impact. The notification must include information on the proposed activity

and its expected impacts, as well as an invitation to express their interest to participate in the decision making process. If an interest is expressed, relevant information regarding the EIA must be provided.

Aarhus Convention and the UNECE *Convention on Access to Information, Public Participation in Decision making and Access to Justice in Environmental Matters* (known as theAarhus Convention), signed on 25 June 1998 by 39 countries and by the European Community. It will come into force after receiving the 16th ratification.
The Aarhus Convention is based upon three principles:

- Access to environmental information
- Public participation
- Access to justice.

The focus here is on public participation, although access to information and access to justice are key elements in effective public participation processes. The Aarhus Convention sets stricter provisions for public participation than those of the EIA Directive; the Directive will thus need to be amended in order to comply with the Aarhus Convention to which the EU is a signatory. The provisions for public participation in EIA apply to the permitting process of any activity listed in Annex I of the Convention, including the decommissioning of nuclear reactors.

3.8 Funding Requirements

Decommissioning of nuclear facilities is a process that requests important financial resources; moreover the project covers a long period of activities that, including the long-term management of nuclear waste and spent fuel, can reach some decades. Special concern has to be devoted to guarantee that the financial burden can be sustained by operators. It would be most inconvenient to extinguish economical and financial resources when work is not finished and, perhaps, is not in a safe state.
Regulators intend therefore establish specific rules in order to guarantee that sufficient resources are always available; this happens with the constitution of decommissioning specific funds, and with rules regarding the management of the funds themselves.
The funds for decommissioning are generally collected by the operator, who raises the money by means of its commercial activities, namely by means of a levy on the price of the sold kWh during the operational lifetime of the plant. As payments for gradual decommissioning may be required over long periods of time, the amount of funds collected at the time of the final shutdown should be calculated taking into account the discounted decommissioning costs.

The level of regulatory control over collection and management of the decommissioning funds varies considerably between different states. In some countries the management of the accumulated reserves remain entirely in the hands of the plant operator, whereas in other countries the funds have to be put under control of a governmental agency.

At European level, a work is now in progress in order to define a specific Directive on the collection and on the management of the decommissioning funds.

3.9 References

[3.1.] NEA/IAEA/EC Workshop on The Regulatory Aspects of Decommissioning. 19-21 May, 1999 – ANPA - Rome, Italy.

[3.2.] The Convention on Early Notification of a Nuclear Accident (Adopted 26 September 1986)

[3.3.] The Convention on Assistance in the case of a Nuclear Accident or radiological Emergency (Adopted 26 September 1986)

[3.4.] The Convention on nuclear safety (Adopted 17 June, 1994)

[3.5.] The Joint Convention on the safety of Spent Fuel Management and on the safety of Radioactive Wastes. The Joint Convention entered into force on 18 June 2001 and as of December 2001 had been ratified by 27 IAEA Member States.

[3.6.] Treaty establishing the European Atomic Energy Community (signed in Rome on 25 March 1957)

[3.7.] EC 1999/829/EURATOM Commission Recommendation of 6 December 1999 on the application of Article 37 of the Euratom Treaty.

[3.8.] EU Council Directive 96/29/Euratom - 13 May 1996 (Basic Safety Standards)

[3.9.] EU Council Directive 97/11/EC of 3 March 1997 amending Directive 85/337/EEC on the assessment of the effects of certain public and private projects on the environment

[3.10.] EC 98/C 133/3 Communication from the Commission 30 April 1998

[3.11.] IAEA Safety Series 115 – 1996 (International Basic Safety Standard for Protection against Ionizing Radiation and for the Safety of Radiation Sources).

[3.12.] IAEA Safety Series 111-F – 1995 The principles of radioactive Waste Management

[3.13.] IAEA Safety Guides WS-G-2.1 – 1999: Decommissioning of Nuclear Power Plant and Research Reactor

[3.14.] IAEA Safety Guides WS-G-2.2 – 1999: Decommissioning of Medical, Industrial and Research Facilities

[3.15.] IAEA Safety Guides WS-G-2.4 – 2001: Decommissioning of Nuclear Fuel Cycle Facilities

[3.16.] IAEA Safety Requirements WS-R-2 – 2000: Decommissioning of Nuclear Fuel Cycle Facilities

[3.17.] ICRP Publication 60 – 1990 Recommendations of the International Commission on Radiological Protection

[3.18.] ICRP Publication 77 - 1997: Radiological Protection Policy for the Disposal of Radioactive Waste

[3.19.] ICRP Publication 81 - 1999: Radiation Protection Recommendations as Applied to the Disposal of Long-lived Solid Radioactive Waste.

[3.20.] IAEA Safety Series 89 – 1988: Principles for the Exemption of Radiation Sources and Practices from Regulatory Control

[3.21.] IAEA Safety Series 111-P-1.1 – 1992: Application of Exemption to the Recycle and Reuse of Materials from Nuclear Facilities: a Safety Practice

[3.22.] IAEA Safety Series 111-G-1.5 – 1995: Clearance Levels for Radionuclides in Solid Materials

[3.23.] IAEA TECDOC-855 (Interim report for comment) 1996: : Clearance Levels for Radionuclides in Solid Materials.

[3.24.] EUR 18860 -1999: Decommissioning of nuclear installations in the European Union

[3.25.] EC Radiation Protection 89 – 1998: Recommended radiological protection criteria for the recycling of metals from the dismantling of nuclear installations.

[3.26.] EC Radiation Protection 113 – 2000: Recommended radiological protection criteria for the clearance of buildings and building rubble from the dismantling of nuclear installation.

[3.27.] EC Radiation Protection 114 – 1999: Definition of clearance levels for the release of radioactively contaminated buildings and building rubble Final Report.

[3.28.] EC Radiation Protection 122 – 2000: Practical use of the concepts of clearance and exemption.

[3.29.] IAEA TECDOC-1097 1999: Maintenance of Records for Radioactive Waste Disposal.

[3.30.] IAEA Safety Series 111-G-1.1 – 1994 Classification of Radioactive Waste.

[3.31.] EC 1999/669 Commission Recommendation of 15 September 1999 on a classification system for solid radioactive waste

[3.32.] IAEA Safety Series 116 – 1994: Design of Spent Fuel Storage Facilities

[3.33.] IAEA Safety Series 117 – 1994: Operation of Spent Fuel Storage Facilities

[3.34.] IAEA Safety Series 118 – 1994: Safety Assessment for Spent Fuel Storage Facilities

[3.35.] IAEA Safety Series TS-R-1 (ST-1, Revised) – 2000: Regulations for the Safe Transport of Radioactive Material

Nuclide	IEX	KCL	E01	E10	E150	ICL	ICA	IST	IAL	IBS	IBV	IEQ	JCm	JCM	GUC	GCC
Ag-110m	1E1	--	7.4E-3	7.4E-2	7.4E-1	3E-1	--	--	--	--	--	--	--	--	--	--
Am-241	1E0	--	2.96E-3	2.96E-2	3.7E-1	3E-1	3E-1	1E0	9E-1	2E-1	3E-1	7E-1	1E-1	1E0	--	--
C-14	1E4	1E2	3.7E0	3.7E1	7.4E2	3E2	--	--	--	--	--	--	1E1	1E2	2E2	2E2
Co-58	1E1	--	--	--	--	3E0	--	--	--	--	--	--	--	--	--	--
Co-60	1E1	--	3.7E-3	3.7E-2	7.4E-1	3E-1	1E-1	3E-1	3E-1	1E-1	1E0	1E1	1E-1	1E0	1E-1	2E0
Cs-134	1E1	--	1.48E-2	1.48E-1	2.22E0	3E-1	--	--	--	--	--	--	1E-1	1E0	--	--
Cs-137*	1E1	--	3.7E-2	3.7E-1	7.4E0	3E-1	5E-1	1E0	1E0	4E-1	4E0	4E1	1E-1	1E0	5E-1	7E0
Fe-55	1E4	--	3.7E2	3.7E3	7.4E4	3E2	1E4	1E3	2E5	9E1	9E2	5E3	1E3	1E4	3E3	1E4
Fe-59	1E1	1E2	--	--	--	3E0	--	--	--	--	--	--	--	--	--	--
H-3	1E6	1E2	--	--	--	3E3	--	--	--	--	--	--	1E1	1E2	3E2	3E2
I-129	1E2	--	3.7E-3	3.7E-2	7.4E-1	3E1	--	--	--	--	--	--	1E-1	1E0	4E-2	4E-2
I-131	1E2	1E2	--	--	--	3E0	--	--	--	--	--	--	--	--	--	--
Mn-54	1E1	--	1.85E-2	1.85E-1	2.96E0	3E-1	4E-1	1E0	1E0	4E-1	4E0	4E1	1E0	1E1	--	--
Na-24	1E1	1E2	--	--	--	3E-1	--	--	--	--	--	--	--	--	--	--
Ni-59	1E4	--	7.4E2	7.4E3	1.11E5	--	--	--	--	--	--	--	1E3	1E4	--	--
Np-237*	1E0	--	2.59E-3	2.59E-2	3.7E-1	3E-1	--	--	--	--	--	--	--	--	--	--
Pu-239	1E0	--	3.7E-3	3.7E-2	7.4E-1	3E-1	3E-1	1E0	9E-1	2E-1	3E-1	7E-1	1E-1	1E0	4E-2	1E0
Ra-228*	1E1	--	1.11E-2	1.11E-1	1.48E0	3E-1	--	--	--	--	--	--	--	--	--	--
Sr-89	1E3	--	--	--	--	3E2	--	--	--	--	--	--	--	--	--	--
Sr-90*	1E2	--	7.4E-3	7.4E-2	1.11E0	3E0	5E1	2E2	3E2	1E1	7E1	4E1	1E0	1E1	1E1	1E1
Tc-99	1E4	--	1.85E2	1.85E3	2.59E4	3E2	7E3	2E4	5E4	1E3	9E3	5E3	1E0	1E1	4E0	4E0
Tc-99m	1E2	1E2	--	--	--	3E1	--	--	--	--	--	--	--	--	--	--
Th-228*	1E0	--	2.59E-3	2.59E-2	3.7E-1	3E-1	--	--	--	--	--	--	--	--	--	--
Th-230	1E0	--	7.4E-3	7.4E-2	7.4E-1	3E-1	--	--	--	--	--	--	--	--	--	--
Th-232	1E0	--	1.48E-3	1.48E-2	1.85E-1	3E-1	--	--	--	--	--	--	--	--	--	--
U-234	1E1	--	1.11E-2	1.11E-1	1.85E0	3E-1	--	--	--	--	--	--	--	--	--	--
U-235*	1E1	--	1.11E-2	1.11E-1	1.85E0	3E-1	--	--	--	--	--	--	--	--	1E-1	3E0
U-238*	1E1	--	1.11E-2	1.11E-1	1.85E0	3E-1	1E0	5E0	3E0	1E0	1E0	4E0	--	--	1E-1	4E0
Zn-65	1E1	--	3.7E-2	3.7E-1	7.4E0	3E-1	6E-1	2E0	2E0	6E-1	6E0	5E1	1E0	1E1	--	--

- IEX : Exemption Levels (IAEA-SS-115, BSS), Ref. [3]
- KCL : Clearance Levels in Korea
- E01, E10, E150: Draft Clearance Levels for Metal Scrap Recycling at Criteria of 0.1mrem, 1.0mrem, 15mrem per year, respectively (US EPA), Ref. [12]
- ICL : Clearance Levels of IAEA (IAEA-TECDOC-855), Ref. [4]
- ICA, IST, IAL, IBS, IBV, IEQ : Clearance Levels of IAEA for recycling scenarios, concrete, steel, aluminum, surface-contaminated building, volumetric contaminated building, contaminated equipment, respectively (IAEA-SS-111-P-1.1), Ref. [2]
- JCm, JCM : Japanese Draft Clearance Levels, minimum and maximum values, respectively (IAEA-TECDOC-1031)
- GUC, GCC : Unconditional and Conditional Clearance Levels of Germany (SSK Draft Recommendations, 1995)

Table 3.1 – Comparison of various Exception/Clearance levels [Bq/g]

WM'99 Conference, February 28 – March 4, 1999)

DEVELOPMENT OF REGULATORY FRAMEWORK FOR IMPLEMENTING RADIOLOGICAL EXEMPTION/CLEARANCE BASED ON ALARA PRINCIPLES AND CURRENT PARACTICES IN KOREA (Korea Institute of Nuclear Safety)

Contamination limit	Country	Additional Information
0.37 Bq/cm^2	Germany	Over 100 cm^2 for fixed and removable contamination and for each single item
0.40 Bq/cm^2	Finland	Removable surface contamination over 0.1m^2 for accessible surfaces
0.40 Bq/cm^2	Belgium	Mean value for removable surface contamination over 300 cm^2, for beta-gamma emitters and alpha emitters with low radiotoxicity
0.83 Bq/cm^2	USA	Surface contamination above background over no more than 1 m^2, with a maximum of 2.5 Bq/cm^2 above background if the contaminated area does not exceed 100 cm^2
4.00 Bq/cm^2	Sweden	Mean value for removable surface contamination over 100 cm^2, with a maximum of 40 Bq/cm^2 if the contaminated area does not exceed 10 cm^2

Table 3.2 – Surface contamination limits for Beta/Gamma emitters in several countries

Contamination limit	Country	Additional Information
0.10 Bq/g	Germany	- - -
0.10 Bq/g	Sweden	Over and above the content of natural activity that occurs in corresponding goods outside the nuclear installation (primarily for limiting the activity in materials that, having been melted down, can be re-used in new products
0.40 Bq/g	UK	Total activity for solids, other than closed sources, that are substantially insoluble in water
0.40 Bq/ml	UK	Total activity for organic liquids that are radioactive solely because of the presence, either separately or simultaneously, of Carbon 14 and Tritium
1.00 Bq/g	Germany	Re-use of metal in a general melting facility
N/A	USA	The United States has not developed a volumetric release standard

Table 3.3 – Specific cctivity limits regardless of type of emission in several countries

Waste Classes	Typical Characteristics	Disposal Options
Exempted Waste (EW)	Activity levels at or below national clearance levels which are based on an annual dose to members of the public of <0.01 mSv	No radiological restrictions
Low and Intermediate Level Waste (LILW)	Activity levels above clearance levels and thermal power below about 2 kw/m^3	
Short Lived waste (LILW-SL)	Restricted long lived radionuclide concentrations (limitation of long lived alpha emitting radionuclides to 4000 Bq/g 4 GBq/te) in individual waste packages and to an overall average of 400 Bq/g (0.4 GBq/te) per waste package)	Near surface or geological disposal facility
Long Lived waste (LILW-LL)	Long lived radionuclide concentrations exceeding limitations for short lived waste	Geological disposal facility
High Level Waste (HLW)	Thermal power above about 2 kw/m^3 and long lived radionuclide concentrations exceeding limitations for short lived waste	Geological disposal facility

Table 3.4 – Current IAEA solid radioactive waste classification system

Document	Title
IAEA TECDOC-680, Dec 1992	Quality assurance requirements and methods for high level waste packages acceptability
CEC, Report EUR 13069 EN, 1991	Quality assurance in the management of radioactive waste in the European Community
IAEA Technical Report Series N. 350, Jan. 1993	Improved cement solidification of low and intermediate level radioactive wastes
IAEA Technical Report Series N. 376, 1995	Quality assurance for Radioactive Waste Packages
IAEA Technical Report Series N. 399, 2000	Organisation and Management for Decommissioning of Large Nuclear facilities
IAEA TECDOC-1222, Jun 2001	Waste inventory record keeping systems (WIRKS) for the management and disposal of radioactive waste
IAEA Technical Report Series N. 355, 1993	Containers for packaging of Solid Low and Intermediate Radioactive Waste
IAEA Safety Guide N. WS-G-2.1, 1999	Decommissioning of Nuclear Power Plants and Research Reactors

Table 3.5 – Most important documents/guidance dealing with QA Issues

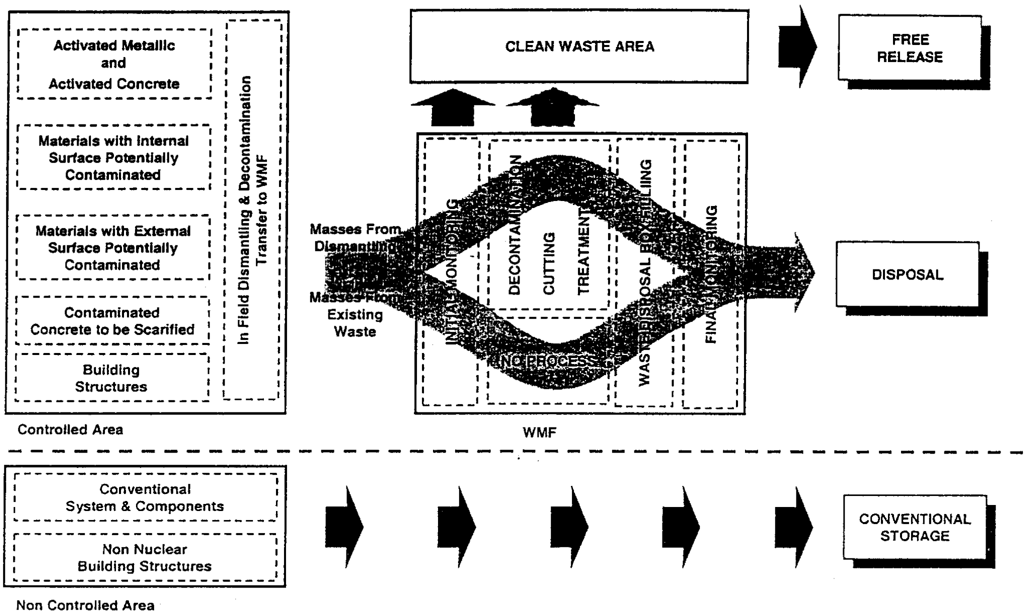

Fig 3.1 – Typical path for material management

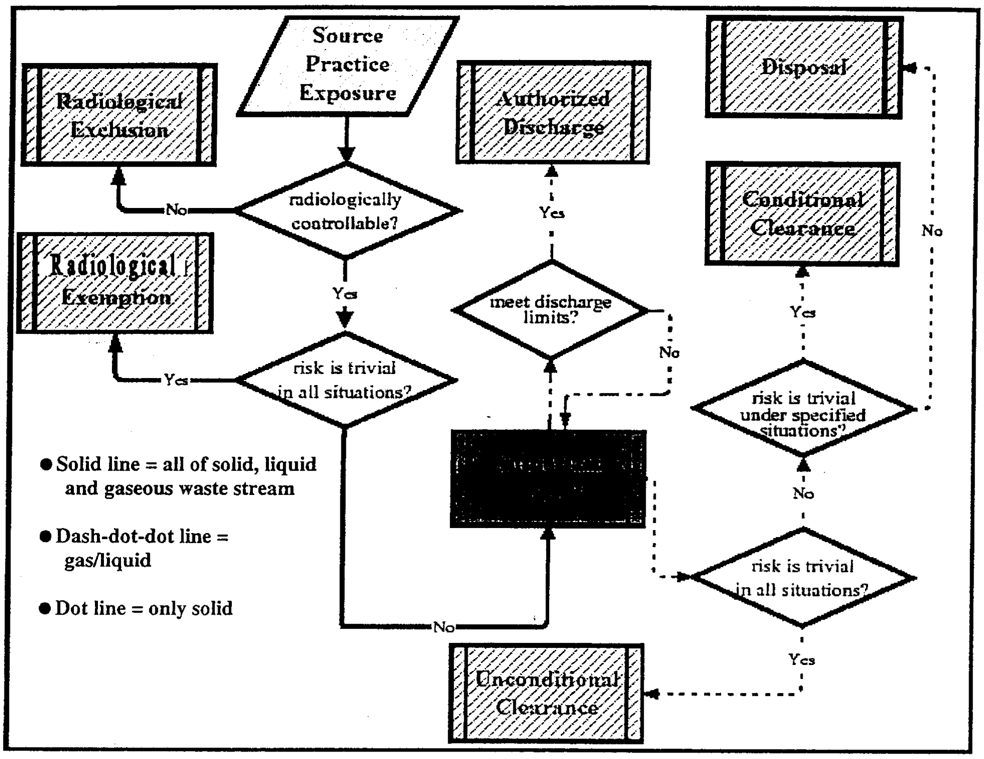

Fig 3.2 – Options for radiation source control

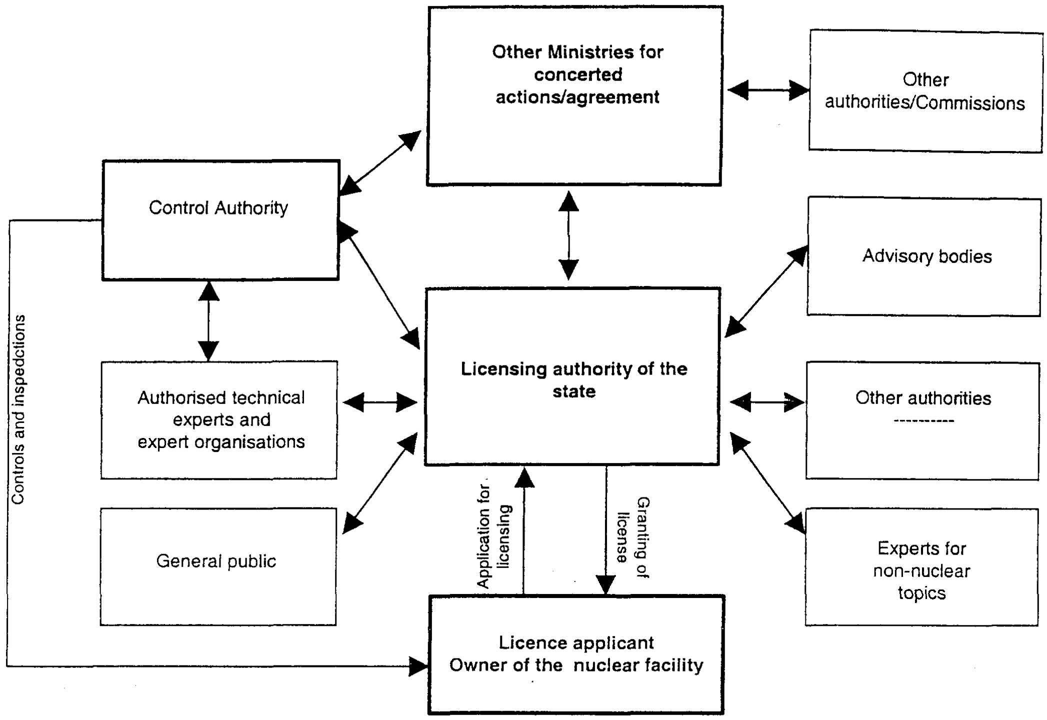

Fig 3.3 – Generic Decommissioning Licence scheme - Players and relationships

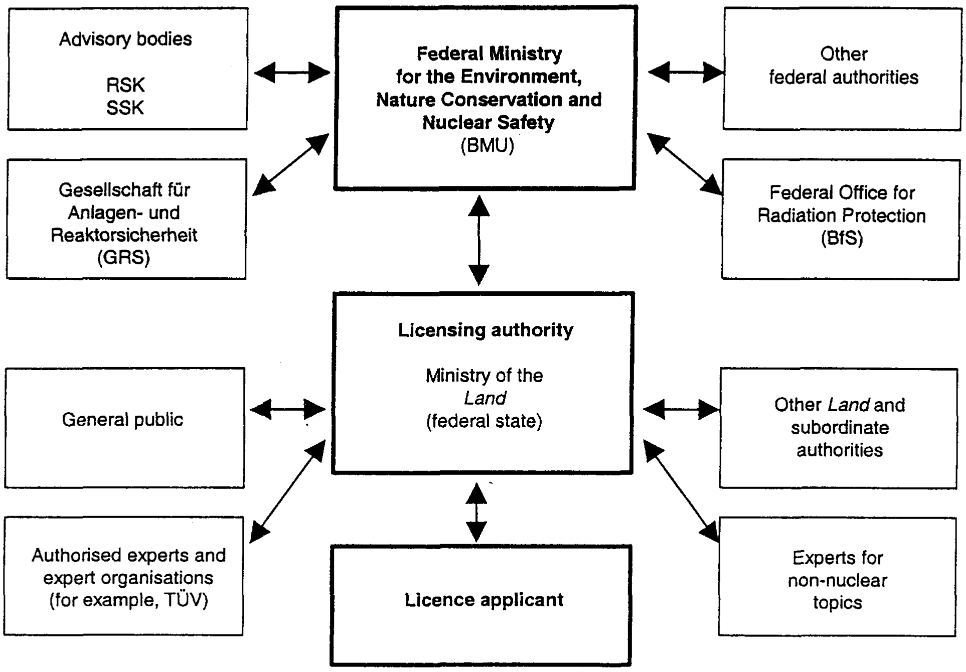

Fig 3.4 – Licence scheme for Germany- Players and relationships

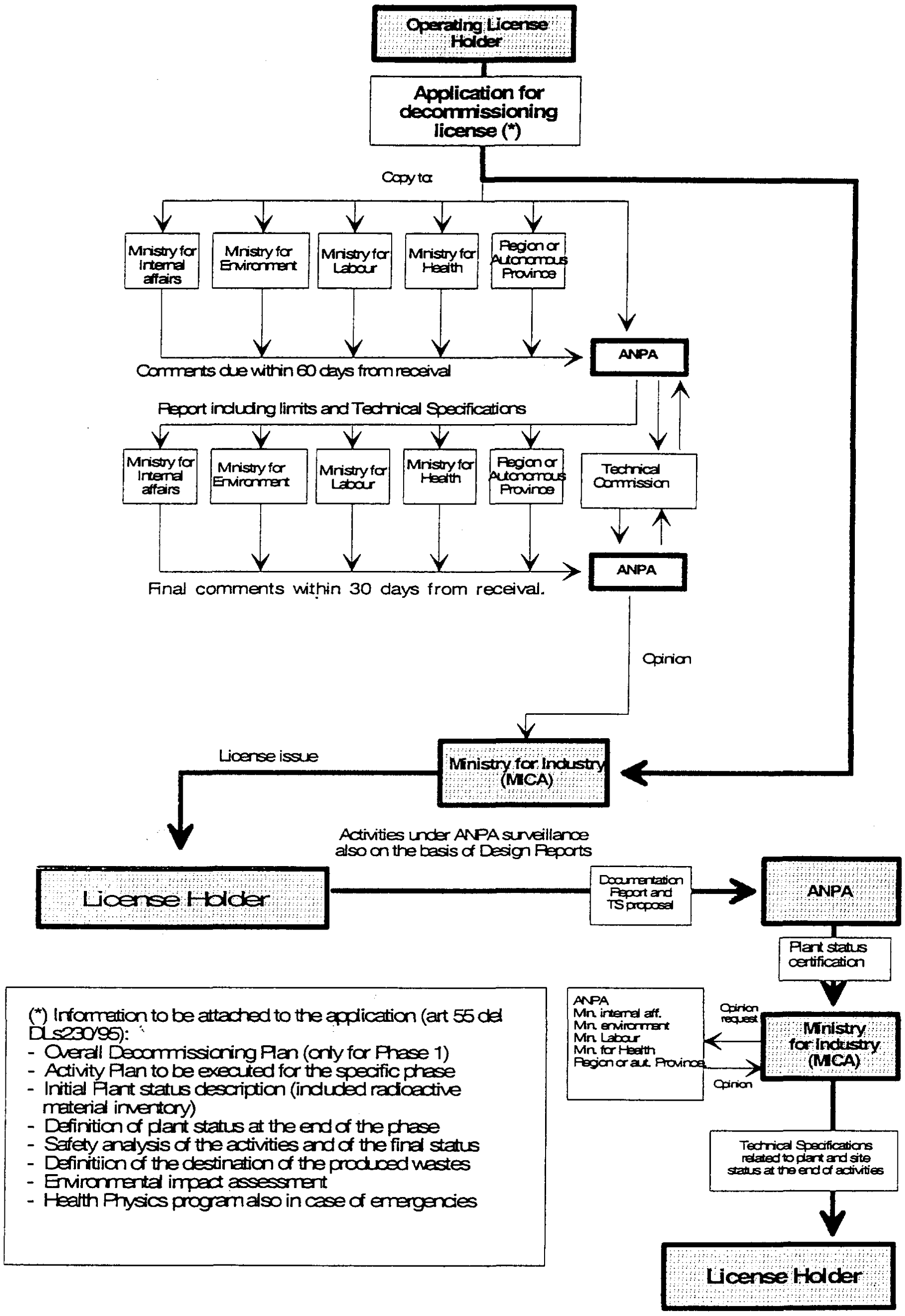

Fig 3.5 – Licence scheme for Italy - Players and relationships

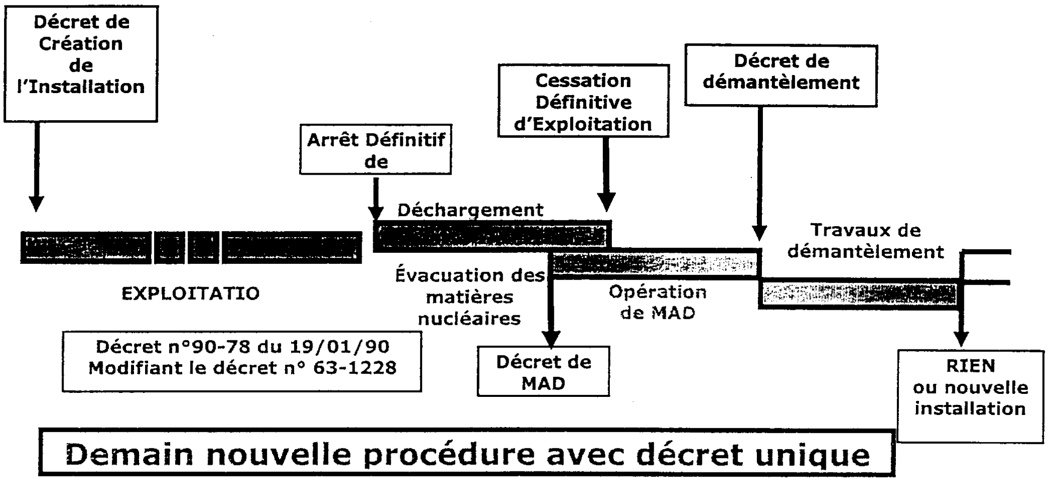

Fig 3.6 – Licence scheme for France

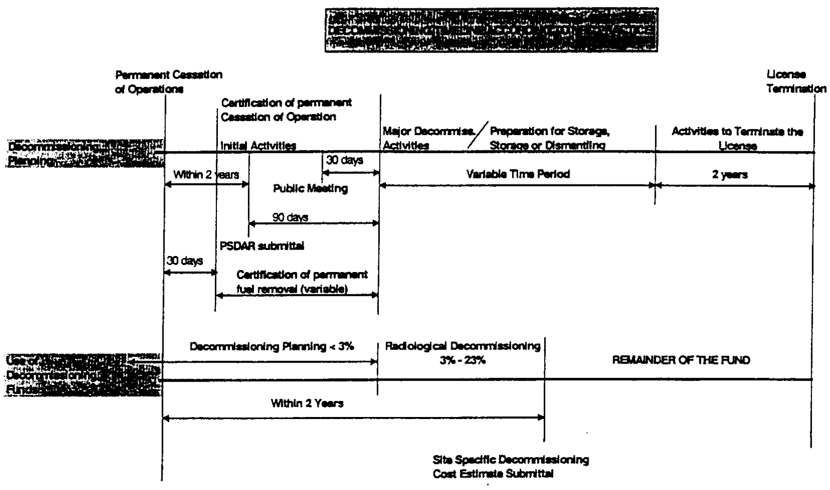

Fig 3.7 – Licence scheme for USA

KEY STAGES	NOTES
Project Preparation	The developer prepares the proposals for the project
Notification to Competent Authority	In some MS there is a requirement for the developer to notify the CA in advance of the application for development consent. The developer may also do this voluntarily and informally.
Screening	The CA makes a decision on whether EIA is required. This may happen when the CA receives notification of the intention to make a development consent application, or the developer may make an application for a Screening Opinion. The Screening decision must be recorded and made public. (See the guidance on Screening in EIA) (Article 4).
Scoping	The Directive provides that developers may request a Scoping Opinion from the CA. The Scoping Opinion will identify the matters to be covered in the environmental information. It may also cover other aspects of the EIA process (see the guidance on Scoping in EIA). In preparing the opinion the CA must consult the environmental authorities (Article 5(2)). In some MS Scoping is mandatory.
Environmental Studies	The developer carries out studies to collect and prepare the environmental information required by Article 5 of the Directive (see Appendix D).
Submission of Environmental Information to Competent Authority	The developer submits the environmental information to the CA together with the application for development consent. If an application for an Annex I or II project is made without environmental information the CA must screen the project to determine whether EIA is required (see above). (Articles 5(1) and 5(3)). In most MS the environmental information is presented in the form of an Environmental Impact Statement (EIS).
Review of Adequacy of the Environmental Information	In some MS there is a formal requirement for independent review of the adequacy of the environmental information before it is considered by the CA. In other MS the CA is responsible for determining whether the information is adequate. The guidance on EIS Review is designed to assist at this stage. The developer may be required to provide further information if the submitted information is deemed to be inadequate.
Consultation with Statutory Environmental Authorities, Other Interested Parties and the Public	The environmental information must be made available to authorities with environmental responsibilities and to other interested organisations and the general public for review. They must be given an opportunity to comment on the project and its environmental effects before a decision is made on development consent. If transboundary effects are likely to be significant other affected MS must be consulted (Articles 6 and 7).
Consideration of the Environmental Information by the Competent Authority before making Development Consent Decision	The environmental information and the results of consultations must be considered by the CA in reaching its decision on the application for development consent (Article 8).
Announcement of Decision	The decision must be made available to the public including the reasons for it and a description of the measures that will be required to mitigate adverse environmental effects (Article 9).
Post-Decision Monitoring if Project is Granted Consent	There may be a requirement to monitor the effects of the project once it is implemented.

The highlighted steps must be followed in all Member States under Directives 85/337/EC and 97/11/EC. Scoping is not mandatory under the Directive but Member States must establish a voluntary procedure by which developers can request a Scoping Opinion from the CA if they wish. The steps which are not highlighted form part of good practice in EIA and have been formalised in some Member States but not in all. Consultations with environmental authorities and other interested parties may be required during some of these additional steps in some Member States.
Abbreviations CA = Competent Authority; MS = Member State.

Fig 3.8 – EIA key stages

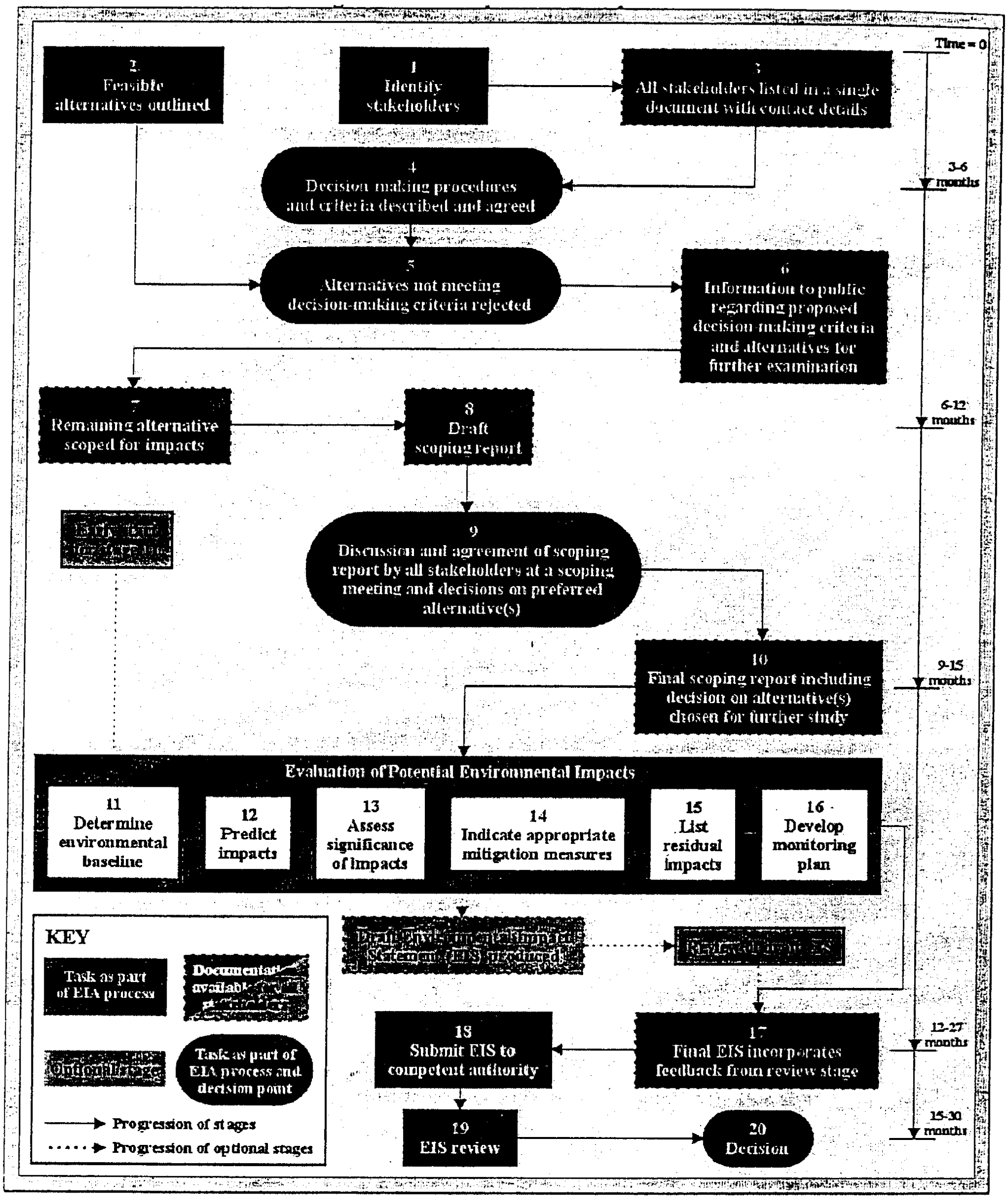

Fig 3.9 – EIA scoping stage details

Annex 3.1

No-Nuclear issues in decommissioning regulation

Besides the specific and dedicated law/regulation for the nuclear facilities decommissioning, also conventional regulations applies to the decommissioning site. They take care of worker safety and health, as well as all other issues related to public health. In general local authorities are in charge of verifying the compliance of the work conditions with national or regional law & regulation dealing with this framework.
The European Agency for Safety and Health at Work has been set up by the European Union in order to serve the information needs of people with an interest in occupational safety and health.
Located in Bilbao (Spain) the Agency has co-ordinated a network since 1997 with Focal Points in each Member State of the Union. The Agency co-operates also with many International Organisations and with safety and health administrations and interested parties world-wide.
The Agency's objective as set out in the founding Regulation is "in order to encourage improvements in the working environment, the Agency shall provide the Community bodies, the Member States and those involved in health and safety at work with the technical, scientific and economic information of use in the field of safety and health at work."

A number of Directives has been established to protect the health and safety of Europe's workers. In addition information on relevant European standards and guidelines related to legislation is also provided in the following.
At present, Community legislation relating to health and safety at work falls into three groups:

- Group 1: measures taken in pursuance of the Framework Directive 89/391/EEC, which contains basic provisions for health and safety organisation at the workplace; it outlines the responsibilities of employers and workers, and is supplemented by individual directives for specific groups of workers, workplaces or substances;
- Group 2: measures taken in pursuance of the Framework Directive 80/1107/EEC, which is designed to protect the health and safety of workers against the risks arising from exposure to chemical, physical and biological agents at the workplace, supplemented by individual directives dealing with specific agents;
- Group 3: measures stemming from directives which contain exhaustive provisions unconnected to the framework directives, in respect of occupational activities or specific groups at risk.

Only the most relevant in the area of decommissioning work are presented.

Group 1:

- **Workplaces**

Council Directive 89/654/EEC of 30 November 1989 concerning the minimum safety and health requirements for the workplace (first individual directive within the meaning of Article 16 (1) of Directive 89/391/EEC). The directive aims to introduce minimum measures designed to improve the working environment, in order to guarantee a better standard of safety and health protection.

- **Use of work equipment**

Council Directive 89/655/EEC of 30 November 1989 concerning the minimum safety and health requirements for the use of work equipment by workers at work (second individual Directive within the meaning of Article 16 (1) of Directive 89/391/EEC). It contains also the Employers' obligations.

- **Use of personal protective equipment**

Council Directive 89/656/EEC of 30 November 1989 on the minimum health and safety requirements for the use by workers of personal protective equipment at the workplace (third individual directive within the meaning of Article 16 (1) of Directive 89/391/EEC). The directive lays down minimum requirements for the assessment, selection and correct use of personal protective equipment. Priority must be given to collective safety measures.

- **Manual handling**

Council Directive 90/269/EEC of 29 May 1990 on the minimum health and safety requirements for the manual handling of loads where there is a risk particularly of back injury to workers (fourth individual Directive within the meaning of Article 16 (1) of Directive 89/391/EEC).

- **Carcinogens**

Council Directive 90/394/EEC of 28 June 1990 on the protection of workers from the risks related to exposure to carcinogens at work (sixth individual Directive within the meaning of Article 16 (1) of Directive 89/391/EEC). The directive lays down minimum requirements for protecting workers against risks arising specifically from exposure to carcinogens and mutagens; to lessen exposure with a view to reducing health risks, to establish exposure limit values and to take preventive measures.
The directive does not apply to workers exposed only to radiation because they are covered by the EAEC Treaty. It applies to workers exposed to asbestos and vinyl chloride monomer when its provisions are more favourable to safety and health at work than are the provisions of the specific directives for those substances.

- **Biological agents**

Directive 2000/54/EC of the European Parliament and of the Council of 18 September 2000 on the protection of workers from risks related to exposure to biological agents at work (seventh individual directive within the meaning of Article 16(1) of Directive 89/391/EEC). The current Directive codifies Directive 90/679/EEC, which has been amended on several occasions since its adoption. The new Directive will supersede the various Directives incorporated in it; their content is fully preserved, and that they are brought together with only such formal amendments as are required by the codification exercise.

- **Chemical agents**

Council Directive 98/24/EC of 7 April 1998 on the protection of the health and safety of workers from the risks related to chemical agents at work (fourteenth individual Directive within the meaning of Article 16(1) of Directive 89/391/EEC). The aims of the directive are to lay down minimum requirements regarding the protection of workers against health and safety hazards related to chemical agents at work or any work activity involving chemical agents. It lays down general principles covering all chemical agents.

- **Temporary or mobile construction sites**

Council Directive 92/57/EEC of 24 June 1992 on the implementation of minimum safety and health requirements at temporary or mobile construction sites (eighth individual Directive within the meaning of Article 16 (1) of Directive 89/391/EEC). To foster an improvement in working conditions in this sector, where workers are exposed to particularly high risks, by taking account of health and safety at the project design and organisation stages. To prevent risks by establishing a chain of responsibility linking all the parties involved. The provisions of the other individual directives apply, with the exception of those of Directive 89/654/EEC on workplaces. This directive well applies to decommissioning workers thought as worker in a temporary site.

Group 2:

- **Asbestos**

Council Directive 83/477/EEC of 19 September 1983 on the protection of workers from the risks related to exposure to asbestos at work (second individual Directive within the meaning of Article 8 of Directive 80/1107/EEC.
Council Directive 91/382/EEC of 25 June 1991 amending Directive 84/477/EEC (second individual Directive within the meaning of Article 8 of Directive 80/1107/EC). The aim of these directives is to establish limit values and specific harmonised minimum requirements for the protection of workers. To reduce exposure to asbestos so as to lessen the risk of diseases occurring.

Group 3:

Besides the above Directives, related to work safety and health, also the following Directives, related to the wastes, have to be taken into account:

- **Hazardous waste**

Council Directive 91/156/EEC of 18 March 1991 amending Directive 75/442/EEC on waste. The general rules applying to waste management which are laid down by Council Directive 75/442/EEC of 15 July 1975 on waste, as amended by Directive 91/156/EEC, also apply to the management of hazardous waste.
Council Directive 91/689/EEC of 12 December 1991 on hazardous waste

All the above Directives have been endorsed (may be in different way) in the national legislations of the member countries.

Annex 3.2

Status of the IAEA Safety Standards Programme(2002-08-26)
September 2002

Legend:
(Blank) No revision planned
□□□□□□ New document or revision planned
▣□□□□□ DPP in preparation or awaiting approval
■□□□□□ DPP approved
■▣□□□□ Document being drafted
■■□□□□ Awaiting approval of Committee(s) for submission to MS
■■▣□□□ Approved by Committee(s)
■■■□□□ Awaiting comments from MS/incorporating comments from MS
■■■▣□□ Awaiting approval by Committee(s) for submission to the Commission
■■■■□□ Endorsed by the Commission/Final editing
■■■■▣□ Submitted to B of G/Publications Committee
■■■■■□ Approved by B of G/Publications Committee
■■■■■▣ In print/in translation
■■■■■■ Published in ....

Throughout this report the first column provides the list of published IAEA Safety Standards. The second column gives the working identification number (DS ...) of standards being developed or revised. **Bold type** indicates standards issued, or to be issued, under the authority of the Board of Governors, others are issued under the authority of the Director General. The last column provides the list of Committees; the first Committee listed has the lead in the preparation and review of that particular standard.

This document is also available at the IAEA Internet site: **www.iaea.org/ns/coordinet**

General Safety

I. Safety Fundamentals

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Remarks / Status / Technical Officer / Expected date of publication	Committee
110: The Safety of Nuclear Installations (1993)	**DS298 Objectives and Principles of Nuclear, Radiation, Radioactive Waste and Transport Safety** (Combining 110, 111-F & 120)	Draft on hold. ■▣□□□□ / Barraclough/	CSS, NUSSC, RASSC, WASSC, TRANSSC
111-F: The Principles of Radioactive Waste Management (1993)	See DS298 (Combining 110, 111-F & 120)		
120: Radiation Protection and the Safety of Radiation Sources (1996) Co-sponsorship: FAO, ILO, OECD/NEA, PAHO, WHO.	See DS298 (Combining 110, 111-F & 120)		

II. Legal and Governmental Infrastructure

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Remarks / Status / Technical Officer/ Expected date of publication	Committee
GS-R-1 Legal and Governmental Infrastructure for Nuclear, Radiation, Radioactive Waste and Transport Safety (2000) Supersedes Safety Series Nos. 50-C-G (Rev. 1) and 111-S-1	DS180	■■■■■■ Published in English / Hughes	NUSSC, RASSC WASSC, TRANSSC
GS-G-1.1 Organization and Staffing of the Regulatory Body for Nuclear Facilities (2002) Supersedes Safety Series No. 50-SG-G1	DS247	■■■■■■ Published in English / Karbassioun.	NUSSC, WASSC
GS-G-1.2 Review and Assessment of Nuclear Facilities by the Regulatory Body (2002) Supersedes Safety Series No. 50-SG-G3	DS248	■■■■■■ Published in English / Karbassioun	NUSSC, WASSC
GS-G-1.3 Regulatory Inspection of Nuclear Facilities and Enforcement by the Regulatory Body (2002) Supersedes Safety Series No. 50-SG-G4 (Rev. 1)	DS289	■■■■■■ Published in English / Karbassioun	NUSSC, WASSC
GS-G-1.4 Documentation for Use in Regulating Nuclear Facilities (2002) Supersedes Safety Series Nos. 50-SG-G8 and 50-SG-G9	DS290	■■■■■■ Published in English / Karbassioun.	NUSSC, WASSC

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General Safety (Cont'd.)

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Remarks / Status / Technical Officer / Expected date of publication	Committee
50-SG-G2 Information to be Submitted in Support of Licensing Applications for Nuclear Power Plants (1979)	DS309 Format and Content of Safety Analysis Reports for NPPs	Submitted to Member States for Comments. Comments are due in September 2002. ■■■□□□ / Ranguelova / 2003 4th qr.	NUSSC, RASSC, WASSC
	DS67 Regulation of Protection and Safety of Radiation Sources in Medicine, Agriculture, Research, Industry and Education. Co-sponsorship: FAO, Int. Labour Office, PAHO & WHO	Submitted to Member States for comments. Comments are due 23 November 2002. ■■■□□□ / Bilbao / 2003 4th qr.	RASSC, WASSC, TRANSSC

III. Emergency Preparedness and Response

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Remarks / Status / Technical Officer / Expected date of publication	Committee
GS-R-2 Preparedness and Response for a Nuclear or Radiological Emergency (2002) Co-sponsorship: FAO, OCHA, OECD/NEA, ILO, PAHO, WHO.	DS43	Endorsed by CSS Nov. 2001. Approved by the Board of Governors in March 2002. In print. ■■■■■▣ / McKenna / 2002 4th qr.	RASSC, NUSSC, WASSC, TRANSSC
50-SG-G6 Preparedness of Public Authorities for Emergencies at Nuclear Power Plants (1982)	DS105 Preparedness for Nuclear and Radiological Emergencies (combining G6, O6 & 98) Co-sponsorship: OECD/NEA, WHO.	Initial review of the draft by Committees in December 2002. ■■□□□□ / McKenna / 2003 4th qr.	RASSC, NUSSC, WASSC, TRANSSC
50-SG-O6 Preparedness of the Operating Organization (Licensee) for Emergencies at Nuclear Power Plants (1982)	See DS105 (combining G6, O6 & 98)		
98 - On-Site Habitability in the Event of an Accident at a Nuclear Facility (1989)	See DS105 (combining G6, O6 & 98)		
109 - Intervention Criteria in a Nuclear or Radiation Emergency (1994)	DS44 Criteria for Use in Planning Response to Nuclear and Radiological Emergencies.	Document being drafted. ■▣□□□□ / McKenna / 2003 4th qr.	RASSC NUSSC, WASSC

General Safety (Cont'd.)

IV. Quality Assurance

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Remarks / Status / Technical Officer Expected date of publication	Committee
50-C/SG-Q Quality Assurance for Safety in Nuclear Power Plants and other Nuclear Installations: Code & Safety Guides Q1-Q14 (1996) ** Q1 Establishing and Implementing a Quality Assurance Programme Q2 Non-conformance Control and Corrective Actions Q3 Document Control and Records Q4 Inspection and Testing for Acceptance Q5 Assessment of the Implementation of the Quality Assurance Programme Q6 Quality Assurance in the Procurement of Items and Services Q7 Quality Assurance in Manufacturing Q8 Quality Assurance in Research and Development Q9 Quality Assurance in Siting Q10 Quality Assurance in Design Q11 Quality Assurance in Construction Q12 Quality Assurance in Commissioning Q13 Quality Assurance in Operation Q14 Quality Assurance in Decommissioning			
	DS113 Quality Management Systems for Regulatory Bodies	Initial review by the Committees in fourth quarter 2002. ■■□□□□ / Boal/ 2003 4th qr.	RASSC, NUSSC, WASSC TRANSSC
	DS314 Quality Management Systems for Technical Services in Radiation Safety	Initial review by the Committees in December 2002. ■■□□□□ / Cruz Suarez/ 2003 4th qr.	RASSC
	DS319 Quality Management Systems in Radiation Safety for Users	Updated DPP to be presented to the Committees in December 2002. ■⊡□□□□ / Dodd/ 2004 2nd qr.	RASSC, TRANSSC

** In the 1996 edition the Code (Requirement) and Guides are in a single document (50-C/SG-Q). The revised safety standards will be published in the General Safety category.

4

Nuclear Safety

I. Design of Nuclear Power Plants

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Status / Remarks / Technical Officer	Committee
NS-R-1 Safety of Nuclear Power Plants: Design (2000) Supersedes Safety Series Nos. 50-C-D (Rev. 1) and 50-SG-D1	DS181	■■■■■■ Published in English / Gasparini	NUSSC
NS-G-1.1 Software for Computer Based Systems Important to Safety in Nuclear Power Plants (2000)	DS261	■■■■■■ Published in English / Hughes	NUSSC
NS-G-1.2 Safety Assessment and Verification for Nuclear Power Plants (2001) Supersedes Safety Series No. 50-SG-D11	DS253	■■■■■■ Published in English / Gasparini	NUSSC
NS-G-1.3 Instrumentation and Control Systems Important to Safety in Nuclear Power Plants (2002) Supersedes Safety Series Nos. 50-SG-D3 and 50-SG-D8	DS252	■■■■■■ Published in English / Pachner	NUSSC
NS-G-1.4 Fuel Handling and Storage Systems in Nuclear Power Plants (2002) Supersedes Safety Series No. 50-SG-D10	DS276	Endorsed by CSS for publication, June 2002. In print. ■■■■■⊡ / Gasparini / 2002 4th qr.	NUSSC
50-SG-D2 (Rev. 1) Fire Protection in Nuclear Power Plants (1992)	DS306 Protection Against Fire and Fire Induced Explosions in Nuclear Power Plants	Submitted to NUSSC for final review in July 2002. CSS review November 2002. ■■■⊡□□ / Tezuka / 2003 3rd qr.	NUSSC
50-SG-D4 Protection against Internally Generated Missiles and their Secondary Effects in Nuclear Power Plants (1980)	DS299 Protection Against Internal Hazards (other than fire and explosions)	Submitted to NUSSC for final review in July 2002. CSS review November 2002. ■■■⊡□□ / Labbé / 2003 3rd qr.	NUSSC
50-SG-D5 (Rev. 1) External Man-induced Events in relation to Nuclear Power Plant Design (1996)	DS301 External Events (excluding earthquakes) in Nuclear Power Plant Design	Resubmitted to NUSSC for final review in June 2002. CSS review November 2002. ■■■⊡□□ / Contri / 2003 3rd qr.	NUSSC
50-SG-D6 Ultimate Heat Sink and Directly Associated Heat Transport Systems for Nuclear Power Plants (1981)	DS282 Design of Reactor Coolant System and Associated Systems in Nuclear Power Plants (combining D6 & D13)	Submitted to NUSSC for final review in July 2002. CSS review November 2002. ■■■⊡□□ / Tezuka / 2003 3rd qr.	NUSSC
50-SG-D7 (Rev. 1) Emergency Power Systems at Nuclear Power Plants (1991)	DS303 Emergency Power Systems at Nuclear Power Plants	Submitted to NUSSC for final review in July 2002. CSS review November 2002. ■■■⊡□□ / Saito / 2003 3rd qr.	NUSSC
50-SG-D9 Design Aspects of Radiation Protection for Nuclear Power Plants (1985)	DS313 Design Aspects of Radiation Protection for Nuclear Power Plants	Initial review by the Committees is expected in October 2002. ■■□□□□ / Gustafsson / 2004 1st qr.	RASSC, WASSC, NUSSC

Nuclear Safety (Cont'd.)

IAEA Safety Standards	Safety Standards in Preparation WID – Title	Status / Remarks / Technical Officer	
50-SG-D12 Design of the Reactor Containment Systems in Nuclear Power Plants (1985)	DS296 Design of Reactor Containment Systems for Nuclear Power Plants	To be issued to NUSSC for final review in October 2002. ■■■⊡□□ / Gasparini / 2003 4th qr.	NUSSC
50-SG-D13 Reactor Coolant and Associated Systems in Nuclear Power Plants (1986)	See DS282 (Combining D6 and D13)		
50-SG-D14 Design for Reactor Core Safety in Nuclear Power Plants (1986)	DS283 Reactor Core Design in Nuclear Power Plants	Issued to NUSSC for final review, comments due 31 July. CSS review November 2002. ■■■⊡□□ / Tezuka / 2003 3rd qr.	NUSSC
50-SG-D15 Seismic Design and Qualification for Nuclear Power Plants (1992)	DS304 Seismic Design and Component Qualification for NPPs	Issued to NUSSC for final review. CSS review in November 2002. ■■■⊡□□ / Contri / 2003 3rd qr.	NUSSC
79 Design of Radioactive Waste Management Systems at Nuclear Power Plants (1986)			
116 Design of Spent Fuel Storage Facilities (1994)			
118 Safety Assessment for Spent Fuel Storage Facilities (1994)			

II. Operation of Nuclear Power Plants

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Remarks / Status / Technical Officer / Expected date of publication	Committee
NS-R-2 Safety of Nuclear Power Plants: Operation (2000) Supersedes Safety Series No. 50-C-O (Rev. 1)	DS179	■■■■■■ Published in English / Vaišnys	NUSSC
NS-G-2.1 Fire Safety in Operation of Nuclear Power Plants (2000)	DS263	■■■■■■ Published in English / Vaišnys	NUSSC
NS-G-2.2 Operational limits and conditions and operating procedures for Nuclear Power Plants (2000) Supersedes Safety Series No. 50-SG-O3	DS185	■■■■■■ Published in English / Vaišnys	NUSSC
NS-G-2.3 Modifications to Nuclear Power Plants (2001)	DS251	■■■■■■ Published in English / Vaišnys	NUSSC
NS-G-2.4 The Operating Organization for Nuclear Power Plants (2001) Supersedes Safety Series No. 50-SG-O9	DS250	■■■■■■ Published in English / Vaišnys	NUSSC

Nuclear Safety (Cont'd.)

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Status / Remarks / Technical Officer	Committee
NS-G-2.5 Core Management and Fuel Handling for Nuclear Power Plants (2002) Supersedes Safety Series No. 50-SG-O10	DS297	■■■■■■ Published in English / Vaišnys	NUSSC
NS-G-2.6 Maintenance, Surveillance and In-Service Inspection in Nuclear Power Plants (2002) Supersedes Safety Series Nos. 50-SG-O2, 50-SG-O7 (Rev. 1) and 50-SG-O8 (Rev. 1)	DS273	Endorsed by CSS for publication, November 2001. In print. ■■■■■⊡ / Vaišnys / 2002 4th qr.	NUSSC
NS-G-2.7 Radiation Protection and Radioactive Waste Management in the Operation of Nuclear Power Plants (2002) Supersedes Safety Series Nos. 50-SG-O5 and 50-SG-O11	DS187	Endorsed by CSS for publication, November 2001. In print. ■■■■■⊡ / Gustafsson / 2002 3rd qr.	RASSC, NUSSC, WASSC
NS-G-2.8 Recruitment, Qualification and Training of Personnel for Nuclear Power Plants (2002) Supersedes Safety Series No. 50-SG-O1 (Rev. 1)	DS287	Endorsed by CSS for publication, June 2002. In print. ■■■■■⊡ / Vaišnys / 2002 4th qr.	NUSSC
NS-G-2.9 Commissioning of Nuclear Power Plants (2002) Supersedes Safety Series No. 50-SG-O4	DS291	Endorsed by CSS for publication, June 2002. In print. ■■■■■⊡ / Vaišnys / 2002 4th qr.	NUSSC
NS-G-2.10 Periodic Safety Review of Nuclear Power Plants (2003) Supersedes Safety Series No. 50-SG-O12	DS307	Endorsed by CSS for publication, June 2002. In print. ■■■■■⊡ / Kotyza&Pachner / 2003 1st qr.	NUSSC
93 System of Reporting Unusual Events in Nuclear Power Plants (1989)	DS288 A National System for Feedback of Experience from Events in NPPs	To be submitted to Member States for comments. ■■⊡□□□ / Hughes / 2003 3rd qr.	NUSSC
117 Operation of Spent Fuel Storage Facilities (1994)			

7

Nuclear Safety (Cont'd.)

III. Site Evaluation for Nuclear Power Plants

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Remarks / Status / Technical Officer / Expected date of publication	Committee
50-C-S (Rev. 1) Code on the Safety of Nuclear Power Plants: Siting (1988)	**DS305 Site Evaluation for Nuclear Facilities**	Submitted to Member States for comments in November 2001, Comment due March 2002. ■■■⊡□□ / Gürpinar / 2003 4th qr.	NUSSC
NS-G-3.1 External Human Induced Events in Site Evaluation for Nuclear Power Plants (2002) Supersedes Safety Series No. 50-SG-S5	DS258	■■■■■■ Published in English / Contri.	NUSSC
NS-G-3.2 Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for Nuclear Power Plants (2002) Supersedes Safety Series Nos. 50-SG-S3, 50-SG-S4, 50-SG-S6 and 50-SG-S7	DS182	■■■■■■ Published in English / Gürpinar	NUSSC
NS-G-3.3 Seismic Hazard Evaluation for Nuclear Power Plants (2002) Supersedes Safety Series No. 50-SG-S1 (Rev. 1)	DS302	Endorsed by CSS for publication, June 2002. In print. ■■■■■⊡ / Gürpinar / 2002 4th qr.	NUSSC
NS-G-3.4 Meteorological Events in Site Evaluation for Nuclear Power Plants (2002) Supersedes Safety Series Nos. 50-SG-S11A and 50-SG-S11B	DS184	Endorsed by CSS for publication, June 2002. In print. ■■■■■⊡ / Labbé / 2002 4th	NUSSC
50-SG-S8 Safety Aspects of the Foundations of Nuclear Power Plants (1986)	DS300 Geotechnical aspects of Nuclear Power Plant Site Evaluation and foundations	Issued to NUSSC for final review in July 2002. CSS review is expected in November. ■■■⊡□□ / Labbé / 2003 3rd qr.	NUSSC
50-SG-S9 Site Survey for Nuclear Power Plants (1984)		Some text to be included in revised Code on Siting 50-C-S (Rev. 1).	
50-SG-S10A Design Basis Flood for Nuclear Power Plants on River Sites (1983)	DS280 Flood Hazard for Nuclear Power Plants on Coastal and River Sites	Review by CSS expected in November 2002. ■■■⊡□□ / Contri / 2003 3rd qr.	NUSSC
50-SG-S10B Design Basis Flood for Nuclear Power Plants on Coastal Sites (1983)	See DS280 (combining S10A and S10B)		NUSSC

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Nuclear Safety (Cont'd.)

IV. Research Reactor Safety

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Remarks / Status / Technical Officer / Expected date of publication	Committee
35-S1 Code on the Safety of Nuclear Research Reactors: Design (1992)	**DS272 Safety of Research Reactors** (Combining 35-S1 and 35-S2)	Submitted to Member States for comments October 01. Comments are due by 15 February 2002. ■■■▣□□ / Boado Magan / 2003 4th qr.	NUSSC
35-S2 Code on the Safety of Nuclear Research Reactors: Operation (1992)	**See DS272** (*Combining* 35-S1 and 35-S2)		
35-G1 Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report (1994)			
35-G2 Safety in the Utilization and Modification of Research Reactors (1994)			
	DS259 Commissioning of Research Reactors	To be re-considered by NUSSC in the light of DS 272. ■■■▣□□ / Boado Magan / 2003 4th qr.	NUSSC
	DS260 Maintenance, Periodic Testing and Inspections of Research Reactors	To be re-considered by NUSSC in the light of DS 272. ■■■▣□□ / Boado Magan / 2003 4th qr.	NUSSC
	DS261 Operational Limits and Conditions for Research Reactors	To be re-considered by NUSSC in the light of DS 272. ■■■▣□□ / Boado Magan / 2003 4th qr.	NUSSC
	DS325 The Operating Organization and the Recruitment, Training and Qualification of Personnel for Research Reactors	DPP approved by CSS in June 2002. ■□□□□□ / Voth / 2004 4th qr.	

V. Safety of Fuel Cycle Facilities

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Remarks / Status / Technical Officer / Expected date of publication	Committee
	DS316 Design and Operation of Non-Reactor Nuclear Facilities	DPP approved, initial draft in preparation. ■▣□□□□ / Nocture /2004 2nd qr.	NUSSC, RASSC, WASSC
	DS317 Safety of Uranium Fuel Fabrication Facilities	DPP approved, initial draft in preparation. ■▣□□□□ / Nocture /2004 2nd qr.	NUSSC, RASSC, WASSC
	DS318 Safety of MOX Fuel Fabrication Facilities	DPP approved. ■□□□□□ / Nocture /2004 4th qr.	NUSSC, RASSC, WASSC

Radiation Safety

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Remarks / Status / Technical Officer / Expected date of publication	Committee
115 International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (1996) Co-sponsorship: FAO, ILO, OECD/NEA, PAHO, WHO. Supersedes Safety Series No. 9, 1982 Edition		■■■■■■ Published in English, Arabic, Chinese, French, Russian and Spanish/	
RS-G-1.1 Occupational Radiation Protection (1999) Co-sponsorship: Int. Labour Office	DS69	■■■■■■ Published in English, Chinese / Gustafsson	RASSC
RS-G-1.2 Assessment of Occupational Exposure due to Intakes of Radionuclides (1999) Co-sponsorship: Int. Labour Office	DS85	■■■■■■ Published in English, Chinese/ Gustafsson	RASSC
RS-G-1.3 Assessment of Occupational Exposure due to External Sources of Radiation (1999) Co-sponsorship: Int. Labour Office	DS12	■■■■■■ Published in English, Chinese/ Gustafsson	RASSC
RS-G-1.4 Building Competence in Radiation Protection and the Safe Use of Radiation Sources (2001) Co-sponsorship: Int. Labour Office, PAHO, WHO	DS73	■■■■■■ Published in English / Bilbao.	RASSC
RS-G-1.5 Radiological Protection for Medical Exposure to Ionizing Radiation (2002) Co-sponsorship: PAHO & WHO	DS22	■■■■■■ Published in English / Ortiz Lopez.	RASSC
26 Radiation Protection of Workers in the Mining and Milling of Radioactive Ores (1983)	DS17 Occupational Radiation Protection in the Mining and Processing of Raw Materials Co-sponsorship: Int. Labour Office	Review by CSS November 2002. ■■■▣□□ / Na / 2003 1st qr.	RASSC
89 Principles for the Exemption of Radiation Sources and Practices from Regulatory Control (1988) Co-sponsorship: OECD/NEA	DS161 Specification of Radionuclide Content in Commodities Requiring Regulation for Purposes of Radiological Protection	Submitted to Member States for Comments. Comments are due 15 September 2002. Next review by the Committees, December 2002. ■■■□□□ / Reisenweaver / 2003 4th qr.	WASSC, RASSC, TRANSSC
	DS114 Safety and Security of Radiation Sources Co-sponsorship: Int. Labour Office, PAHO, WHO	Next review by RASSC expected in December 2002. ■■□□□□ / Dodd / 2003 4th qr.	RASSC
107 Radiation Safety of Gamma and Electron Irradiation Facilities (1992)			

Radioactive Waste Safety

I. Infrastructure

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Remarks / Status / Technical Officer / Expected date of publication	Committee
78 Definition and Recommendations for the Convention on the Prevention of Marine Pollution by Dumping of Wastes and other Matter, 1972 — 1986 Edition			
111-G-1.1 Classification of Radioactive Waste (1994)			
	DS62 Strategy for Environmental and Source Monitoring for Public Protection Purposes	DPP approved, initial draft in preparation. ■⊡□□□□ / Balonov /	RASSC, WASSC

II. Pre-disposal

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Remarks / Status / Technical Officer / Expected date of publication	Committee
WS-R-2 Predisposal Management of Radioactive Waste, including Decommissioning (2000)	DS152	■■■■■■ Published in English / Linsley	WASSC
WS-G-2.1 Decommissioning of Nuclear Power Plants and Research Reactors (1999) Supersedes Safety Series Nos. 74 and 105	DS257	■■■■■■ Published in English, Chinese/ Reisenweaver	WASSC, NUSSC
WS-G-2.2 Decommissioning of Medical, Industrial and Research Facilities (1999)	DS173	■■■■■■ Published in English / Reisenweaver	WASSC
WS-G-2.3 Regulatory Control of Radioactive Discharges to the Environment (2000) Supersedes Safety Series No. 77	DS25	■■■■■■ Published in English / Linsley	RASSC, WASSC
WS-G-2.4 Decommissioning of Nuclear Fuel Cycle Facilities (2001)	DS171	■■■■■■ Published in English / Reisenweaver	WASSC, NUSSC
WS-G-2.5 Predisposal Management of Low and Intermediate Level Radioactive Waste (2003)	DS159	Endorsed for publication November 2001. In Print. ■■■■■⊡ / Metcalf / 2003 1st qr.	WASSC, NUSSC
WS-G-2.6.Predisposal Management of High Level Radioactive Waste (2002)	DS163	Endorsed by CSS for publication November 2001. In Print. ■■■■■⊡ / Metcalf / 2002 4th qr.	WASSC, NUSSC
108 Design and Operation of Radioactive Waste Incineration Facilities (1992)			
	DS160 Management of Radioactive Waste from the Use of Radioactive Materials in Medicine, Industry and Research	Review by Committees in December 2002, before submission to CSS. ■■■⊡□□ / Metcalf / 2003 4th qr.	WASSC, RASSC

Radioactive Waste Safety (Cont'd).

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Remarks / Status / Technical Officer / Expected date of publication	Committee
	DS284 Safety Assessment for Nuclear and Radiation Facilities other than Reactors and Waste repositories	Review by Committees in December 2002, before submission to CSS. ■■■⊡□□ / Metcalf / 2003 4th qr.	WASSC, NUSSC, RASSC
	DS292 Storage of Radioactive Waste	To Be submitted to Member States for comments in 2002. ■■⊡□□□ / Metcalf / 2004 2nd qr.	WASSC, NUSSC

III. Disposal

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Remarks / Status / Technical Officer / Expected date of publication	Committee
WS-R-1 Near Surface Disposal of Radioactive Waste (1999) Supersedes Safety Series No. 96	DS153	■■■■■■ Published in English / Linsley	WASSC
WS-G-1.1 Safety Assessment for Near Surface Disposal of Radioactive Waste (1999)	DS166	■■■■■■ Published in English, Chinese / Linsley	WASSC
WS-G-1.2 Management of Radioactive Waste from the Mining and Milling of Ores (2002) Supersedes Safety Series No. 85	DS277	Endorsed by CSS November 2001. In Print. ■■■■■⊡ / Reisenweaver / 2002 4th qr.	WASSC
111-G-3.1 Siting of Near Surface Disposal Facilities (1994)			
111-G-4.1 Siting of Geological Disposal Facilities (1994)			
99 Safety Principles and Technical Criteria for the Underground Disposal of High Level Radioactive Wastes (1989)	**DS154 Geological Disposal of Radioactive Waste** Co-sponsorship: OECD/NEA	Draft in preparation. ■⊡□□□□ / Metcalf / 2004 4th qr.	WASSC

IV. Rehabilitation

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Remarks / Status / Technical Officer / Expected date of publication	Committee
	DS162 Cleanup of Areas Contaminated by Past Activities and Accidents	Reviewed by the Committees in December 2002. Review by CSS expected in June 2003. ■■■⊡□□ / Reisenweaver / 2003 3rd qr.	WASSC, RASSC
	DS172 Cleanup of Areas Contaminated by Past Activities and Accidents	Draft in preparation. ■⊡□□□□ / Reisenweaver / 2004 1st qr.	WASSC, RASSC

Transport Safety

IAEA Safety Standards	Safety Standards in Preparation WID - Title	Remarks / Status / Technical Officer / Expected date of publication	Committee
TS-R-1 Regulations for the Safe Transport of Radioactive Material (2000) Supersedes ST-1 (1996), Safety Series No. 6 (1985 Edition) and No. 80		■■■■■■ (Published in English, French, Russian, and Spanish). Errata for F, R, and S Edition being prepared/ Dicke	TRANSSC
TS-G-1.1 Advisory Material for the Regulations for the Safe Transport of Radioactive Material (2002) Supersedes Safety Series Nos. 7 and 37	DS245	■■■■■■ Published in English / Pope	TRANSSC
TS-G-1.2 Planning and Preparing for Emergency Response to Transport Accidents Involving Radioactive Material (2002) Supersedes Safety Series No. 87	DS246	Endorsed by CSS (Nov. 1999, May 2001). In Print. ■■■■■⊡ / Pope / 2002 2nd qr.	TRANSSC
	DS326 Quality Assurance for the Safe Transport of Radioactive Material To supersede Safety Series No.113	DPP approved by CSS in June 2002. Initial draft in preparation. ■⊡□□□□ / Bernard-Bruls / 2003 4th qr.	TRANSSC RASSC

Annex 3.3

List of IAEA Documents Published Since 1985 on Strategic and Technical Aspects of Decommissioning of Nuclear Facilities

	Technical Reports Series (TRS)	
1.	Decontamination of Nuclear Facilities to Permit Operation, Inspection, Maintenance, Modification or Plant Decommissioning	TRS No. 249 (1985)
2.	Methodology and Technology of Decommissioning Nuclear Facilities	TRS No. 267 (1986)
3.	Methods for Reducing Occupational Exposure During the Decommissioning of Nuclear Facilities	TRS No. 278 (1987)
4.	Decontamination and Demolition of Concrete and Metal Structures During the Decommissioning of Nuclear Installations	TRS No. 286 (1987)
5.	Factors Relevant to the Recycling or Reuse of Components Arising from the Decommissioning and Refurbishment of Nuclear Facilities	TRS No. 293 (1988)
6.	Monitoring Programmes for Unrestricted Release Related to Decommissioning of Nuclear Facilities	TRS No. 334 (1992)
7.	Cleanup and Decommissioning of a Nuclear Reactor After a Severe Accident	TRS No. 346 (1992)
8.	Application of Remotely Operated Handling Equipment in the Decommissioning of Nuclear Facilities	TRS No. 348 (1993)
9.	Planning and Management for the Decommissioning of Research Reactors and Other Small Nuclear Facilities	TRS No. 351 (1993)
10.	Decontamination of Water Cooled Reactors	TRS No. 365 (1994)
11.	Decommissioning Techniques for Research Reactors	TRS No. 373 (1994)
12.	Safe Enclosure of Shutdown Nuclear Installations	TRS No. 375 (1995)
13.	Design and Construction of Nuclear Power Plants to Facilitate Decommissioning	TRS No. 382 (1997)
14.	Decommissioning of Nuclear Facilities Other than Reactors	TRS No. 386 (1998)
15.	Radiological Characterization of Shutdown Nuclear Reactors for Decommissioning Purposes	TRS No. 389 (1998)
16.	State-of-the-Art Technology for Decontamination and Dismantling of Nuclear Facilities	TRS No. 395 (1999)
17.	Organization and Management for Decommissioning of Large Nuclear Facilities	TRS No. 399 (2000)
18.	Methods for the Minimisation of Radioactive Waste from Decontamination and Decommissioning of Nuclear Facilities	TRS No. 401 (2001)
19.	Record keeping Guidelines and Experience for Decommissioning of Nuclear Facilities	TRS No.411 (2002)
20.	The Transition from Operation to Decommissioning of Nuclear Installations	(under preparation)
	Technical Documents (TECDOC)	
1.	Decontamination and Decommissioning of Nuclear Facilities: Final Report of Three Research Meetings (1984-87)	IAEA-TECDOC 511 (1989)
2.	Decontamination of Transport Casks and of Spent Fuel Storage Facilities	IAEA-TECDOC-556 (1990)
3.	Factors Relevant to the Sealing of Nuclear Facilities	IAEA-TECDOC-603 (1991)
4.	Considerations in the Safety Assessment of Sealed Nuclear Facilities	IAEA-TECDOC-606 (1991)
5.	National Policies and Regulations for Decommissioning Nuclear Facilities	IAEA-TECDOC-714 (1993)
6.	Decontamination and Decommissioning of Nuclear Facilities - Results of a Co-ordinated Research Programme, Phase II: 1989-1993	IAEA-TECDOC-716 (1993) Tecdoc 755
7.	New Methods and Techniques for Decontamination in Maintenance or Decommissioning Operations - Results of a Co-ordination Research Programme, 1994-1998	IAEA-TECDOC-1022 (1998)
8.	Technologies for Gas Cooled Reactor Decommissioning, Fuel Storage and Waste Disposal, Proceedings of a Technical Committee Meeting held in Juelich, Germany, 8-10 September 1997	IAEA-TECDOC-1043 (1998)
9.	On Site Disposal of Nuclear Facilities as a Decommissioning Strategy	IAEA-TECDOC-1124 (1999)
10.	The Decommissioning of WWER-type Nuclear Power Plants	IAEA-TECDOC-1133 (2000)
11.	Nuclear Graphite Waste Management - TCM held in Manchester UK 18-20 Oct 1999	CD-ROM (2001)
12.	Decommissioning Techniques for Research Reactors –Final report of a co-ordinated research project 1997-2001	IAEA-TECDOC-1273 (2002)
13.	The Decommissioning of Small Medical, Industrial and Research Facilities	(approved for publication)
	Nuclear Data Series (NDS)	
1.	Nuclear Data Requirements for Fission Reactor Decommissioning	INDC (NDS)-269 (1993)
2.	International Benchmark Calculations of Radioactive Inventory for Fission Reactor Decommissioning	INDC (NDS)-355 (1996)
	Other	
1.	A Proposed Standardised List of Items for Costing Purposes in the Decommissioning of Nuclear Installations - Interim Nuclear Document	OECD/NEA, Paris (1999)
2.	Joint NEA/IAEA/EC Workshop on the Regulatory Aspects of Decommissioning, 19-21 May 1999, Rome, Italy	ANPA, Rome (2000)

Annex 3.4

EU legislation specific to Transport of Radioactive

Council Directive 89/618/Euratom: Information of the general public about health protection measures to be applied and steps to be taken in the event of a radiological emergency.
This Directive is intended to define, at Community level, common objectives with regard to measures and procedures for informing the general public for the purpose of improving the operational health protection provided in the event of a radiological emergency.

Council Directive 92/3/Euratom: Supervision and control of shipments of radioactive waste between Member States and into and out of the Community.
This Directive shall apply to shipments of radioactive waste between Member States and into and out of the Community whenever the quantities and concentration exceed the levels laid down in Directive 80/836/Euratom. Specific provisions concerning reshipment of such waste are set out in Title IV.

Commission decision n°93/552: Standard document for the supervision and control of shipments of radioactive waste referred to in Council Directive 92/3/Euratom.
In this decision a standard document is defined which shall be used in respect of any shipments of radioactive waste between Member States or into and out of the Community within the scope of Directive 92/3/Euratom.

Council regulation n°1493/93: Shipment of radioactive substances between Member States.
This Regulation establishes a monitoring of transports of sealed sources and other relevant sources between Member States, to replace controls suppressed in 1993 at the internal borders of the EU. It also applies to shipments of radioactive waste, between Member States, as covered by Directive 92/3/Euratom.
In the case of nuclear materials, each Member State carries out all necessary controls, within its own territory, in order to ensure that each consignee of such materials, which are the subject of a shipment from another Member State, complies with the national provisions implementing of Directive 96/29/Euratom.

Council Directive 94/55/CE: Approximation of the laws of the Member States with regard to the transport of dangerous goods by road.
ADR's regulations are applicable to international transports; this Directive makes them applicable also to national transports. The Directive, however, does not apply to the transport of dangerous goods by vehicles belonging to or under the responsibility of the armed forces.

Council Directive 96/35/EC: Appointment and vocational qualification of safety advisers for the transport of dangerous goods by road, rail and inland waterway.
The Member States shall take the necessary measures in accordance with the requirements of this Directive to ensure that no later than 31 December 1999, undertakings the activities of which include the transport or the related loading or unloading, of dangerous goods by road, rail or inland waterway, each appoint one or more safety advisers for the transport of dangerous goods, responsible for helping to prevent the risks inherent in such activities with regard to persons, property and the environment.

Council Directive 96/49/CE: Approximation of the laws of the Member States with regard to the transport of dangerous goods by rail.

RID's regulations are applicable to international transports; this Directive makes them applicable also to national transports. This Directive does not apply, however, to of dangerous goods conducted by means of transport equipment belonging to or under the responsibility of the armed forces.

Council Directive 2000/18/EC: Minimum examination requirements for safety advisers for the transport of dangerous goods by road, rail or inland water.
This Directive lays down the minimum examination requirements for the examination needed to obtain the EC certificate of training as safety adviser for the transport of dangerous goods provided for in Directive 96/35/EC. Member States shall take all necessary measures to ensure that safety advisers for the transport of dangerous goods are examined in such a way that they satisfy these minimum requirements.

Annex 3.5

10 Golden Rules
-lessons learned by EWN-

1. The applicant presents the project and the licensing strategy to the authority before the application will be made and also consults the authority.
2. The applicant confines to the relevant licensing aspects in the licensing procedure and submits, as far as possible, complete, checkable and consistent documents.
3. The applicant defines the start and goal situation oriented on the protection aims and thus, reduces the necessary efforts for authorized opinions.
4. The participants in the procedure (operator, licensing authority and experts) send all papers in parallel to all other persons involved so that time-consuming detours can be excluded and everybody will have the same information level.
5. The participants in the procedure agree on a definite time schedule. The licensing authority binds its expert to check the documents on completeness and to list the additional requirements in an appropriate time.
6. The participants in the procedure regularly organize status discussions to explain the status of the procedure, to define weak points and to specify time scheduling.
7. The licensing authority invites to technical discussions. Technical discussions are also allowed between applicant and expert; the authority decides in the individual case on the participation.
8. The participants in the procedure read the expert opinion draft together to clear up differences of opinions as fast as possible.
9. The authority only itemizes the relevant documents in the granted licence.
10. The participants agree on the necessary details and timely integration of the required documents necessary for the fulfilment of the additional requirements.

Appendix 1

REGULATORY ASPECTS OF DECOMMISSIONING

THE LEGISLATIVE AND NORM BASIS ON REGULATION OF DECOMMISSIONING OF NUCLEAR INSTALLATIONS IN RUSSIA

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1. The main assumptions of the legislative basis on regulation of decommissioning of nuclear installations

The legislative regulation on decommissioning of NPP units in Russia is based on assumptions of:

- the Federal Law "About use of the atomic energy" /1/
- the Federal Law "Radiation safety of population" /2/,
- the Law of Russian Federation "About conservation of environment" /3/

and some assumptions of other legislative deeds, likely to influence on establishing of the lawful relations in the field of use of atomic energy.

The Federal Law "About use of the atomic energy" have been admitted in October 20, 1995. It determines the lawful basis and principles of regulations of relations, appeared during use of the atomic energy. This Law embeds the fundamentals for the creation of system of lawful and norm regulations of activity in the field of the atomic energy, including decommissioning of NPP power units. The priority is fixed for the Laws of Russian Federation and the federal norms and regulations.

This Law determines the follow objects of its implementation:

- the nuclear installations,
- the radiation sources,
- the sites of storage of nuclear materials and radioactive substances,
- the repositories of radioactive waste and other nuclear and radiation-hazardous material

at all stages of their life cycle, including decommissioning.

The Federal Law "About use of the atomic energy" determines that all the nuclear installations, having no defense mean, including NPP, are the subjects of the federal propriety. That is why the decision about their decommissioning are admitted by the Government of the Russian Federation.

This Law states that the exploiting organizations are fully responsible for the safety of the nuclear installations, including their decommissioning and radioactive waste and nuclear material management.

To execute these functions the exploiting organizations have to possess an appropriate power, material and other resources, including financial ones, which are sufficient for the realization of their functions. For these purposes, the exploiting organizations together with the appropriate institutions on management of use of the atomic energy obliged to create the special fund for the decommissioning of NPP units.

This Law determines that any activity in the field of use of the atomic energy has to be carried out only with the presence of the licence for the rights to carry out the appropriate works in the field of use of the atomic energy. These licences are issued by the authorities of the safety state regulation and delivered to the organizations, which exploit, carry out and offer the services in the field of use of the atomic energy.

The Federal Law "About radiation safety of population" has been admitted on December 1995. It states the lawful fundamentals for the provision of radiation safety for population with a purpose to preserve public health.

This Law determines

- the main principles of radiation safety provision,
- the main sanitary norms of exposure (permissible doses),
- the measures required for the radiation safety provision.

The main principles of radiation safety provision include:

- the principle of norm - the non-exceeding of the permissible limits of individual irradiation doses of citizens from all the sources of ionizing radiation;
- the principle of approval - the prohibition of all types of activity on use of ionizing radiation, at which the resulted advantage for individuals and society does not exceed the risk of possible damage due to exposure above the natural radiation background.
- the principle of optimization – the maintenance of individual irradiation doses and number of exposed persons on the possibly low and achieved level during use of any sources of ionizing radiation, regarding the economical and social factors.

This Law determines that since the January 1, 2000 the follow main sanitary norms of exposure (permissible limits of doses) are introduced on the territory of the Russian Federation:

- for the population – the average annual effective dose is equal to 0.001 sievert, or, the effective dose for the period of life (70 years) is equal to 0.07 sievert; during some years the higher values of the effective dose are permitted under condition that the average annual effective dose, will not exceed 0.001 sievert, while accounted for the five successive years
- for the employers - the average annual effective dose is equal to 0.02 sievert, or, the effective dose for the labor period (50 years) is equal to 1sievert; the exposure is permitted up to the annual effective dose as 0.05 sievert under condition, that the average annual effective dose, will not exceed 0.02 sievert, while accounted for the five successive years

The given regulated values of the main limits of irradiation doses do not include the doses,

- resulting from the natural radiation and technogenically changed radiation background,
- being received by the citizens (patients) during medical roentgen-radiological procedures and treatment.

The Law of the Russian Federation "About conservation of environment" entered in force on December 19, 1991. Its last edition entered in force on December 20, 2001. This Law together with measures of organizational, lawful, economical and educational orientation, is called up to promote the forming and consolidation of ecological lawful arrangement and provision of ecological safety on the territory of the Russian Federation and Republics of the Russian Federation.

This Law determines the general ecological requirements during:

- the arrangement, technical-economical justification and planning of the projects,
- the creation, reconstruction and start up of enterprises, installations and other objects, including NPP,
- their decommissioning.

During realization of mentioned types of productive activity the execution of the requirements has to be provided on conservation of natural environment, regarding the near and far ecological, economical, demographical and moral consequences of mentioned activity.

The technical-economical justification of all projects on activity in the field of use of the atomic energy and the projects themselves have to pass through the state expertise, as well as the public and ecological ones if needed.

The projects are not submitted for the approval, if they do not meet ecological requirements. The works on their realization have not to be funded.

This Law determines the obligation of full compensation for the damage, caused by ecological infringement of the law, as well as procedure and scale of such compensation.

2. The main norm document requirements to decommissioning of nuclear installations

The norms and rules on regulation of activity in the field of atomic energy, operated in Russia till the present time, still in a great deal success the system of the norm documents of the former USSR, where no law about use of the atomic energy existed.

In connection with this the operated in Russia norms and rules on regulation of decommissioning of nuclear installations, in particular, of NPP units, are mainly based on the statements of norm documents admitted in nuclear-power complex of the former USSR. Some statements of these norm documents need to be revised in a full accordance with the Federal Laws of the Russian Federation.

At the present time Russia has no complete norm base on regulation of decommissioning of nuclear installations and still does not possess the real full-scale practical experience concerns. The main statements of norm documentation, presently regulated the decommissioning procedure are mainly based on

- experience in repairing and dismantling, accumulated on the operated NPP units, industrial, research and transport nuclear reactors,
- results of scientific research, fulfilled on nuclear installations, which were shut down.

The basic norm and guiding documents of Russia determined the main statements on regulation of decommissioning procedure, for NPP units, in particular, are

- "The main statements on provision of safety of Nuclear Power Plants" (OPB-88/97) /4/ and
- "The regulations on provision of safety during decommissioning of Nuclear Power Plant unit" (NP-012-99) /5/,

entered in force on September 1, 2000.

In accordance with requirements of these norm documents, the decommissioning of NPP unit has to be taken into account as already during its projecting and exploitation, as technical maintenance and repairs.

During 5 years prior end of NPP unit planned operation period, the exploiting organization has to develop, justify and submit the NPP unit decommissioning Program to the Gosatomnadzor of Russia (State Atomic Control Body).

Prior NPP unit decommissioning, the complex (engineering and radiation) inspection has to be done by the commission, appointed by exploiting organization. Based on its results the exploiting organization has to develop the NPP unit decommissioning project and prepare report on justification of decommissioning safety.

The requirements OPB-88/97 and NP-012-99 determines that prior the removal of nuclear fuel, the NPP unit has to meet the exploitation requirements including that to the volume of technological maintenance and number of personnel, occupied with exploitation.

As was mentioned, for realization of NPP unit decommissioning the exploiting organization has to obtain the license from the Gosatomnadzor of Russia, and the NPP unit decommissioning project developed has to pass through ecological (state and public) expertise.

If the conditions of provisions NPP unit decommissioning safety does not correspond to the admitted norm limits and does not provide ecological safety, the works on decommissioning have not to be allowed and the license on this type of activity has not to be issued.

The estimation of impact on environment during realization of the project has to be included in the set of documents of NPP unit decommissioning project.

It is necessary also to notice that different specific aspects of NPP units decommissioning, including the problems of safety provision, are not yet concretized in the present norm documents, operated in Russia.

In particular, the requirement is absent on justification of decommissioning duration in case of postponed dismantling for the stage of long-term conservation of reactor under control.

The possibility of NPP units decommissioning is not taken into consideration in case of immediate dismantling.

Besides this, the existing norm requirements do not account the difference of design peculiarities of different types of reactors, exploiting on the NPP in Russia. Such factors as the NPP multiunity with construction in a line with 2 units are not accounted. As result the locations of some systems and the building are common for the units of one line.

Till the present time the conception, criteria and requirements are not elaborated in Russia for management and re-use of low-level radioactive material (limited and unlimited use), generated during NPP units decommissioning in a significant amount.

The absence of suitable repositories or burials for radioactive waste of NPP enforces to plan their long-term storage on the NPP sites. Nevertheless no requirements are still determined to the organization with storage site and to the procedure of long-term storage of radioactive waste on the NPP sites.

No conception of management of irradiated graphite still was formulated for decommissioning of uranium-graphite nuclear reactors.

In connection with above mentioned, the practice is still used when the obtaining of appropriate permission from Gosatomnadzor of Russia is required in each concrete case to execute some kind of works on decommissioning of nuclear power installations, which was already shut down.

At the present time the Gosatomnadzor of Russia undertake efforts towards development of unified norm base on regulation and concretizing of different aspects of activity dealing with decommissioning of nuclear installations of different purpose.

It is supposed that the experience accumulated in different countries will be taken into consideration for the forming of this norm base.

3 The peculiarities of the procedure for obtaining of licence on decommissioning of nuclear installations

Since July 17, 1977, in Russia, the statement about licensing of activity in the field of use of the atomic energy has been approved and entered in force, including decommissioning of nuclear installations /6/.

In this document the general requirements are established to the

- order of application submission,
- procedure of its evaluation drawing up and
- decision on issue of licenses on different types of activities in the field of use of the atomic energy.

In accordance with the Federal Law "About use of the atomic energy" the follow activities which require the licence, were determined by the Government of the Russian Federation for the organizations or enterprises of any forms of proprietary:

1. The creation, start up of exploitation, exploitation, decommissioning of nuclear installations, radiation sources and sites of storage of nuclear materials and radioactive waste.
2. The nuclear materials and radioactive substances management.
3. The radioactive waste (including radioactive materials of secondary use) management during their storage, treatment, transportation and disposal.
4. The carrying out of scientific-research and experimental-design works with use of the atomic energy and/or use of nuclear materials and radioactive substances.
5. The realization of safety exploitation and technological processes, provision of branch (production) safety control.
6. The projecting and construction of nuclear installations, radiation sources and sites for storage of nuclear materials and radioactive waste.
7. The construction and creation of equipment for nuclear installations, radiation sources and sites for storage of nuclear materials and radioactive waste.
8. The expertise of projecting (including project on decommissioning of nuclear installations) and other materials and documentation with a goal to determine the level of safety of nuclear installations, radiation sources and sites for storage of nuclear materials and radioactive waste.

As follows from this list, the exploiting organization has to obtain the licence, issued by Gosatomnadzor of Russia to realize such type of activity as decommissioning of nuclear installations.

The licence on rights to realize that certain kind of activity is the official document, which allows to realize the determined kind of activity during determined period under determined conditions.

During process of application evaluation and decision on delivery of licences on decommissioning of nuclear installations, the Gosatomnadzor of Russia co-ordinates its activities with federal authorities of executive power and authority bodies of subject of the Russian Federation.

For the obtaining of licences on decommissioning of nuclear installations, of NPP unit, in particular, the exploiting organization (applicant) has to submit to the Gosatomnadzor of Russia the application, which includes:

- The decision of Government of the Russian Federation about decommissioning of NPP;
- the application with indication of organization-lawful form of legal entity, juridical address, kind of activity and object of its implementation;
- the document, which confirms the agreement on this kind of activity between executive authority of subject of the Russian Federation and appropriate authority of the local self-government, on which territory the realization of applied kind of activity is supposed;
- the conclusion of state ecological expertise;
- the set of documents, which justify the application.

The application submitted to the Gosatomnadzor of Russia is accepted for consideration or declined, following the results of preliminary control. During such a control, the nomenclature of submitted document is checking, including the fulfillment of established rules of its drawing up.

During process of consideration of application the Gosatomnadzor of Russia carry out, on its own decision, the inspection of NPP unit to be decommissioned, expertise the safety justification, contact applicant on question of elimination of remarks and disadvantages, occurred.

The decision on delivery or deny of delivery of license is taking by the chiefs of the Gosatomnadzor of Russia on the base of expert conclusion and results of inspection.

The follow documents are included in the set, which justify the application on obtaining of license on NPP unit decommissioning, determined by requirements RD-04-27-2000) /7/:

1. The Program of decommissioning of NPP unit.
2. The report on results of complex (radiation and engineering) inspection of NPP unit.
3. The report on justification of safety during decommissioning. of NPP unit.
4. The Program and graphic of works on dismantling of equipment, systems and protective building constructions.
5. The Program of quality provision during decommissioning of NPP unit.
6. The instruction on liquidation of accidents on NPP unit under decommissioning.
7. The plan of measures on safety of personnel in the case of accident on NPP unit during its decommissioning.
8. The instructions on exploitation of equipment and systems which take into account the stages of dismantling of equipment and system, following dismantling program.
9. The NPP unit decommissioning project material, following list, agreed with

Gosatomnadzor of Russia.

10. The instructions on accounting and control of radioactive waste generated during decommissioning of NPP unit.

11. The inquiry about changing in the systems of physical safety dealing with decommissioning.

12. The Act about carrying out of physical inventory of nuclear materials on zones of balance of materials, which is the last on the moment of submission of application.

13. The document which confirms the absence of nuclear materials on NPP unit with indication of time of nuclear materials exportation.

It is necessary, nevertheless, to notice that the obligatory requirements produced by Gosatomnadzor of Russia towards content of the mentioned documents are not yet established in full volume.

It is supposed that these requirements will be elaborated in full volume and entered in force in the nearest future.

Appendix 2

2002

TEXT-BOOK "EUNDETRAF" (JCAT contribution)

INTRODUCTION

SOME DIRECTIONS OF RUSSIAN INTEREST IN COLLABORATION ON DECOMMISSIONING OF NUCLEAR AND RADIATION HAZARDOUS INSTALLATIONS (NRHI)

Case of RRC "Kurchatov Institute" Moscow. Russia

The following directions in decommissioning of nuclear and radiation hazardous installations could be of interest in collaboration.

- The organization and realization of joint projects on exchange of expertise
- The joint analysis and overview of these experiences
- The implementation of world-wide experience in the Russian conditions, including technical and economical ones
- The joint realization of the multi-faceted decommissioning Projects, namely, dealing with
    - dosimetry and radiation safety of personnel during dismantling works accounting the design specificity of research facilities and complexes (case of RRC "Kurchatov Institute")
    - evaluation of risk under accident scenario at decommissioning works in closed location to the urban zone
    - organization of environmental monitoring and provision of ecological safety
    - big-scale works on dismantling of radioactive equipment and on management of generated radioactive waste
    - analysis of possible transfer of radionuclides in the environment (ground waters, etc...)
    - radiological aspects.

The issue on the problems and tasks, dealing with NRHI decommissioning planning and preparation, regarding the multi-formity of NRHI and site specificity (as the closed location to the urban zone), dictate the necessity in complex approach and scrupulous multi-faceted planning to search for and implement an adequate solution.

The organization and realization of the common Projects could include the follow steps:

- study of principles on creation of informational data base, regarding that in the Western countries, on technical, economical and other aspects of the decommissioning planning and realization; including the preparation of NRHI to the decommissioning (case RRC "Kurchatov Institute")

- study of methodical instruments for the NRHI decommissioning planning and its common implementation on concrete sites in Russia, regarding the western experience
- common development of complex program of step-by-step NRHI decommissioning, regarding the abovementioned data base
- common development of the set of standard documentation on concrete NRHI decommissioning, required to obtain the license of the Gosatomnadzor of Russia, namely:
  - NRHI decommissioning program
  - Program on complex engineering and radiation monitoring of concrete NRHI
  - Decommissioning quality provision program
  - Report on justification of NRHI decommissioning safety
  - Report on evaluation of environmental impact of NRHI decommissioning.

The Russian JCAT, the scientific corporation of Russian enterprises, is interested in collaboration with foreign partners having a practical experience and expertise on nuclear decommissioning, including the possibility of on-site training (stage) in NRHI decommissioning planning and realization for Russian specialists.

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5. Federal norms and rules "Rules on provision of safety during decommissioning of NPP unit" (NP-012-99) M.: 2000.
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7. The requirements to the set and the content of documents justified the provision of nuclear and radiation safety of nuclear installations, storage site, radiation sources and/or applied activity (for NPP) (PD-04-27-2000), Annex 17-guiding document of Gosatomnadzor of Russia, M.: 2001.

EUNDETRAF

Chapter 4

Project Management

Chapter summary

4.1. Introduction

4.1.1 Background Information

Decommissioning project management is a manifold task. The diverse array of project responsibilities and activities require wide familiarity with, and understanding and working knowledge of nuclear engineering and managerial skills. These attributes are necessary to assure that projects achieve planned technical and safety objectives and are accomplished on schedule, within cost and scope. The high cost value, complexity, and longevity of a large size decommissioning project demand a well-structured management approach, prescribed authorities and responsibilities, and finite planning and control. Those generally accepted management practices will be described and serve as a common basis to identify the peculiarities related to decommissioning of nuclear facilities. Where necessary special aspects of the management for decommissioning nuclear facilities will be considered. Practical examples and case studies will complete the text where appropriate.
The Chapters 4.2 to 4.9 comprise common accepted areas of Project Management focussing on but not limited to:

(1) Definition of a decommissioning strategy
(2) Clear definition of the work scope (WBS)
(3 An integrated schedule, including milestones.
(4) A management team in charge and accountable
(5) An understanding of responsibility and authority
(6) Overall Cost estimate and supportive budget planning and execution
(7) Project execution against the plan
(8) Project Management Information System (PMIS)
(9) Periodic management review to ascertain project status
(10) Quality Assurance in decommissioning practice
(11) Risk management
(12) Contract and Procurement Management

4.1.2 Management Processes in Decommissioning

Project Management in general could be defined as:

Planning, organising, controlling and directing of resources (money, materials, time and people) to accomplish a clear defined objective.

Management of nuclear projects shall protect workers, public and environment from both radiological and non-radiological hazards resulting from the activities performed. This requirement presents a remarkable input in management of decommissioning projects throughout the whole project life and results in significant impact on the project implementation and performance. Key issues specific to the decommissioning of nuclear facilities are described e.g. in chapter 2 of the IAEA Safety Guides No. WS-G-2.1-4 [4.1].
"The term 'decommissioning' refers to administrative and technical actions taken to allow the exemption of some or all of the regulatory controls of a nuclear facility (except a repository which is closed rather than decommissioned)" [4.1].

Different actions are carried out to reduce radiological hazards to the workers, public and environment in accordance with the regulatory requirements and national regulations. These actions involve decontamination, dismantling and removal of radioactive materials, waste, components and structures. They are carried out to achieve a systematic reduction in radiological hazards and are based on preplanning documents and assessments of safety and environmental impact during decommissioning operations.

Project Management in general is accomplished through the use of the following processes [4.2]:

- Initiation
- Planning
- Execution
- Controlling
- Closure.

It is important to acknowledge that these processes are iterative in nature. Planning provides the Execution with a plan, and Execution might provide updates to the original plan as the project progresses. For example, cost estimate of a defined work scope might require a redefinition of several work packages because of unacceptable cost value. Furthermore, during the project life cycle, necessity for changes could appear due to changed boundary conditions.

Initiation of Decommissioning

A decommissioning project of a nuclear facility does not need an initiating process as understood in the classical sense of project management (authorisation for the project). It is prescribed by corresponding laws of nuclear activities and generally specified in the operation licence. Thus, the decommissioning after a certain operation time is legally required and must be performed.

Nevertheless, Initiation will be performed during different project phases as for example during the Management Plan issue. The preparatory work has been described above in ch. 1.

Execution of the Decommissioning Plan

The execution of the project will be performed in accordance with the approved decommissioning plan including the project management plan and supplementary licensing application documents. Precondition for executing any task is an obtained licence or an agreement by the authority for the implementation. The required level is depending on safety relevance impact of the task to be performed. Main management attention in executing the decommissioning should be focused on:

- Decommissioning Plan Execution
- Quality Assurance
- Personal
- Information System
- Contracting and administration

The coordination of the work will be performed at the work order (WO) implementation level. The engineering preparation of these detailed WOs might present a huge amount of project management workload depending of the project stage. This is for example the case for the dismantling stage of commercial NPP or the phase of preparation of the safe enclosure.

Controlling of Decommissioning Projects

Controlling the decommissioning project is a substantial part of the management work. The main areas of project control consist of:

- Overall Decommissioning Plan Control
- Schedule Control
- Cost Control
- Quality Control
- Risk Monitoring and Control.

In today's management practice, project control is understood primarily as a quality assurance process. As such, the decommissioning organisation must assure that the decommissioning plan meets safety and technical performance standards and ensure that the decommissioning in terms of time, cost and resources is performed as close as possible to the plan. For a control system to be meaningful a project information system should be in place. Modern control system must be appropriate for the complexity of the tasks being addressed and the size of the project effort. They should also be timely, simple to employ, and congruent with the events being measured. A possible management information system will be presented below, see chapter 4.7.

Completion of Decommissioning

The final goal of any nuclear decommissioning project is to release the facility or part of it from regulatory control. The status of the removal of controls and remaining restrictions will have to be finally discussed and approved within the context of the approved decommissioning plan. The confirmation of the completion status should be based on complete information on the disposition of waste, materials and premises. Appropriate records must be retained as specified by the regulatory body.

The preparation of a final decommissioning report is recommended by the IAEA [4.1].

4.2. Preparation

The preparatory work has been described in ch. 1 above, the major items being the selection of strategy, project analysis under different boundary conditions and preparation of a technical concept.

4.2.1 Project Management Plan

The Project Management Plan (PMP) aims to set out the objectives, description, methodology, organization, timeline and budget of the project based on the strategy defined. The PMP is the most important document in the overall planning, monitoring, and implementation of a project. It is an approved guide to both project execution and control. The primary aim of the document is to describe the planning assumptions and decisions, facilitate communication among the parties involved, and document approved scope, cost and schedule baselines. Other items of technical, commercial, organisational, personnel, and control issues might be included as well.

The essentials of a PMP are as follows:

- A summary of the project that can be read by anyone in a few minutes and that will provide an understanding of the essentials of the project. It states briefly what has to be done and may mention the methods and techniques to be used.

- A list of milestones, identified in such a way that there can be no ambiguity about whether a milestone has been achieved or not. In general, the number of milestones shall correspond to reasonable portions of the project budget in order to provide adequate monitoring.

- A Work Breakdown Structure (WSB) that is detailed enough to provide meaningful identification of all tasks, plus all higher-level work groupings.

- From the milestone list and the WBS, an activity network that shows the sequence of the elements of the project and how they are related (which ones can be done concurrently, which can start only when another is finished, etc.). This is clearly more useful than just marking end points on bar charts.

- Separate budgets and schedules for all the elements of the project for which a certain individual is responsible.

- An interface plan that shows how the project communicates with stakeholders, most particularly with the contractor, but also with staff organizations that are involved.

- An indication of the reporting and review process, what reports are needed, who reviews the project, when, and for what purpose.

- A list of key project personnel and their assignments in relation to the WBS. Key personnel are those responsible for the various phases of the project.

4.2.2 Work Breakdown Structure (WBS)

A fundamental aspect of effective project planning is the process of defining the scope of the project and of breaking it down into manageable pieces of work (Work Packages). The WBS is a task oriented detailed breakdown, which defines the work or tasks to be performed. The WBS initiates the development of the Organisational Breakdown Structure (OBS), and the Cost Breakdown Structure (CBS). Thus, the WBS is the primary planning and analysis tool used in almost all projects, because it answers two questions:

- What is to be accomplished?
- What is (are) the necessary hierarchical relationship(s) of the work effort?

The WBS also aids the project management process by:

- Providing a complete survey of work tasks that must be performed,
- Defining the responsibilities, personnel, cost, duration, risk, and
- Providing an easy-to-follow numbering system to allow hierarchical tracking of progress.

Thus, the WBS partitions the project into manageable elements of work, for which costs, budgets, and schedules can more readily be established.

The formation of the WBS (family tree) begins with the subdivision of the project objective into smaller work blocks until the lowest level to be supported and controlled is reached. This treelike structure breaks down the project work scope into manageable and independent units that are assigned to the various experts responsible for their completion. The WBS links the company resources with the work to be performed.

The detailed breakdown structure is project specific, even a WBS for a similar project can not be directly used. There exists several ways of presenting the organisation of the tasks and processes necessary to complete the project. Whether the WBS organises the project in phases or functional deliverables, at the end of the process the management team should have a common understanding of the project scope.
The following rules in management practice are recognised as relevant:

- Always prepare a WBS.
- Staff a team consisting of project specialists, especially those having a diversity of expertise and experience.
- Be complete. Any element of the project against which funds are expected must be included in the WBS. Make sure that the breakdown is a partition of functional blocks or project activities and not a breakdown by organisation or discipline.
- The WBS is treelike. Therefore, an element at e.g. level 3, must break down into at least two level 4 elements. All of the work performed at the level 3 is performed in level 4.
- Because the structural breakdown is hierarchal, two or more elements in level 2, 3, or 4 can never connect to the same element at a lower level.

After the WBS has been built up, the process will be finalised by establishing the work packages (WP). The WP is a description of what must be performed, by whom, and in what time span. The WP is always prepared for each bottom-level element of the WBS. Thus a level 4 element, if not partitioned, would be ascribed a work package.

After having the WBS established the next step in executing the project plan is to develop a schedule.

4.3. Schedule Management

Planning a project, developing a budget for it, and scheduling all the tasks involved – these three activities are not separable. Planning, budgeting and scheduling are parts of the same basic management process. Because a budget must include both, the amounts and timing of resources expended, one cannot prepare a budget without knowing the specifics of each task and the time periods during which the task has to be undertaken. It is obvious to start the project with the plan development because this planning process is the basis for all subsequent planning activities. The question about whether to deal with first, the budget or the scheduling is arbitrary. Both will be considered separately simply because it is not possible to deal with them at the same time.

Having defined the activities to be performed at the WP level, their possible follow-up and duration will have to be assessed. Based on that input and taking into account resources available, the project schedule can be established in a first issue. The result of that first assessment, depending from various boundary conditions like for example the stakeholders needs, an adjustment process will take place. From this process, finally the initial project schedule will be available. The time management elements will be completed by the schedule control against the project performance achieved. Although, the different elements interact and overlap with each other and the other management processes, they will be presented in separate subchapters because the tools and techniques for each are different.

The project schedule is simply the project plan in an altered format. It is a convenient form for monitoring and controlling project activities. Basically there are 3 types of project schedules:

- Arrow and precedence networks,
- CPM and PERT networks, and
- Simple milestones, bar or Gantt charts.

The most common formats are Gantt charts and PERT/CPM networks that will be used to demonstrate on some examples how to convert a WBS into these formats.

4.3.1 Schedule Development

Based on the dependencies between the activities (network diagram) and the duration estimates performed, it is now possible to establish the project schedule. Completing these two information sets by determining start and finish dates for each of project activities is nothing else than the project schedule development.

Determining start and finish dates will have to take into account also:

- the Risk Management Plan,
- resource availability in terms of time and pattern, calendars conditions (working days, vacation, labour contract limitations, etc.),
- other external influences like licensing process, stakeholders interests, social and budgetary aspects.

Such external impacts might limit the management team's options. Time constraints as "Finish no later than" or " Start no earlier than" are the most commonly used. The identification of key events or major milestones is very important because those are mostly fixed and very difficult to change.

Milestones may be used also to identify new necessary work activities that were not considered in the initial work scope planning. The termination of the fuel removal is an example of an important milestone of any NPP decommissioning project.

4.3.2 Schedule Control

From the discussion above it is clear that a project is running very rarely exactly as scheduled. The schedule is not a monolithic document. During the project, execution changes will appear. Controlling such a living schedule means looking for factors causing deviations from the planning dates, determining changes and managing them. In other words, it must be checked if and to what extend the schedule change will affect work scope, cost, risk, quality, and staffing. Response measures will be taken if milestones from the critical path will be affected. Such corrective action will be taken to bring the activity performance in line with the project plan or ensure the least possible delay.
In complex projects, management software able to support schedule control should be used.

4.4. Cost Management

Beside the discussed technical and time aspects of a decommissioning project there is still one more important issue to be considered - costs. How much is it going to cost? It is a prime concern to get a reasonable and realistic value of the overall cost to have money available when needed. A budget must be developed in order to obtain the resources necessary to accomplish the decommissioning objectives in a safe and optimised manner.

The term "Decommissioning Cost" usually includes all costs from the termination of operation up to the achievement of a "green field " condition at site. However, different approaches are applied and in order to compare costs, it is necessary to check in detail what tasks are included. A proposal of standardised cost item for decommissioning has been published recently [4.3].

4.4.1 Resource Planning

Resource planning means determining what resources (human, equipment, tools, materials) and what quantities of each should be used and when they will be needed. Such planning is primarily based on the WBS and the activity duration estimates.

4.4.2 Cost Estimating

Cost estimating is the process of projecting financial requirements to accomplish the decommissioning objective(s). Although the primary tasks involved can best be described and understood if they are discussed separately and sequentially, in actual practice they are closely related and often carried out concurrently. The primary cost estimating tasks are based on the WBS definition and consist of:

(1) Selecting the estimating structure for preparing cost data.
(2) Collecting, evaluating, and applying the necessary cost and cost related data.
(3) Applying the proper estimating methods.
(4) Documenting the estimate in enough detail, so that it can be reviewed, evaluated, and used in the decision-making process.

When the primary estimate has been completed, uncertainties, limiting assumptions, and constraints should be identified. Changes in basic rules, schedules, quantities, system upgrade, and concepts can significantly affect cost.

There are various techniques for preparing cost estimates. The most frequently used methods are listed below:

- *Top-down estimation*, using the known costs of a similar project as a basis for the current project's cost estimate. It is frequently used in the early project phases for estimating total project costs when details are not completely known.

- *Expert Opinion Technique*. May be used for certain parts of the overall project when other techniques or data are not available. Several specialists can be consulted reiteratively until a consensus on the cost estimate is established.

- *Bottom-Up estimation*. Generally, a summation of the costs for individual work packages or activities used to get finally the total project costs.

- *Parametric Technique*. Parametric estimation requires historical data on similar projects. Statistical analysis is performed on the data to find correlations between cost drivers and other system parameters, such as performance parameters. The analysis produces cost equations or cost estimating relationships that can be used individually or grouped into more complex models.

- *Trend Analyses Technique*. An efficiency index is derived by comparing originally projected contract costs against actual costs of work performed to date. The index is used to adjust the cost estimate of work not yet completed.

- *Cost Review and Update Technique.* An estimate is performed by examining previous estimates of the same project for internal logic, completeness of work scope assumptions and estimating methodology. The estimates are then updated to reflect the cost impact of new conditions.

When a cost estimate is prepared for a project, a description of the basis for the cost estimate shall be made and included in the documentation. The general requirements for cost estimates can be summarized as follows:

(1) Planning/Feasibility Study Estimate. The basis for the cost estimate shall comprise a description of the project's purpose, new facilities and special tools, significant features and components, proposed methods of accomplishment, proposed decommissioning schedule, and any other pertinent cost positions.

(2) Budget Estimate. These cost estimates shall be based on all the detailed requirements in the decommissioning concept report such as the performance parameters, applicable codes, specifications and standards. Quality assurance requirements, methods of performance, operations interfaces, safety requirements, licensing process and so forth, should be considered.

(3) Enhanced Estimates. The basis for these cost estimates shall include all the approved engineering data, methods of performance, final project parameters, project schedule, and final exact detailed requirements.

4.4.3 Budgeting

A budget is simply a plan in terms of cost for allocating resources to the project activities.
The project planning process has been described above, as a set of steps that began with the overall project plan and then divided and subdivided the plan's elements into smaller and smaller pieces that could finally be sequenced, assigned, scheduled and cost estimated. Hence, the project budget is nothing else than the project plan, based on the activities plan or WBS, expressed in monetary terms.
Budgeting the project should involve some contingencies to be able to manage unexpected changes during the project performance. This risk management area will be discussed in some more details below in ch. 4.8.

Once the budget has been established, it acts as a tool for the upper management to monitor and guide the project. Appropriate data must be collected and reported in a timely manner. This collection and reporting system must be carefully designed in the initial project plan, in order to avoid late and inaccurate reporting, that could negate the main purpose of the budget.

4.4.4 Cost Control

Control of costs is achieved through monitoring, analysing, reporting and exercising control over commitments and expenditures with due regard to schedule. A key element is maintaining visibility of the forecast final cost of the project and exercising corrective action in good time, by regular progress assessment and determination of future commitments and expenditures.

One of the important elements of the cost control is the implementation of an adequate accounting system. Project accounting deals with the control and historic recording of actual cash payments within the project organisation itself as well as outside the own organisation as for example to consultants, contractors, suppliers provided to the project in a manner which enables comparison with the project budget.

Definitive and detailed procedures are essential for this controlling function to ensure the financial integrity and transparency of the project.

4.5. Quality Management

Quality assurance (QA) begins at the project conception stage and runs through all decommissioning project phases. QA affects cost, availability, effectiveness, safety and impact on the environment. Therefore, QA aspects should be given careful consideration during the project preparation and performance.

4.5.1 Elements of QA Management

The quality requirements and the QA activities considered necessary to accomplish the project objectives must be laid down in the QA Handbook and QA procedures. Consideration shall be given to the following elements for their appropriate inclusion in a QA program.

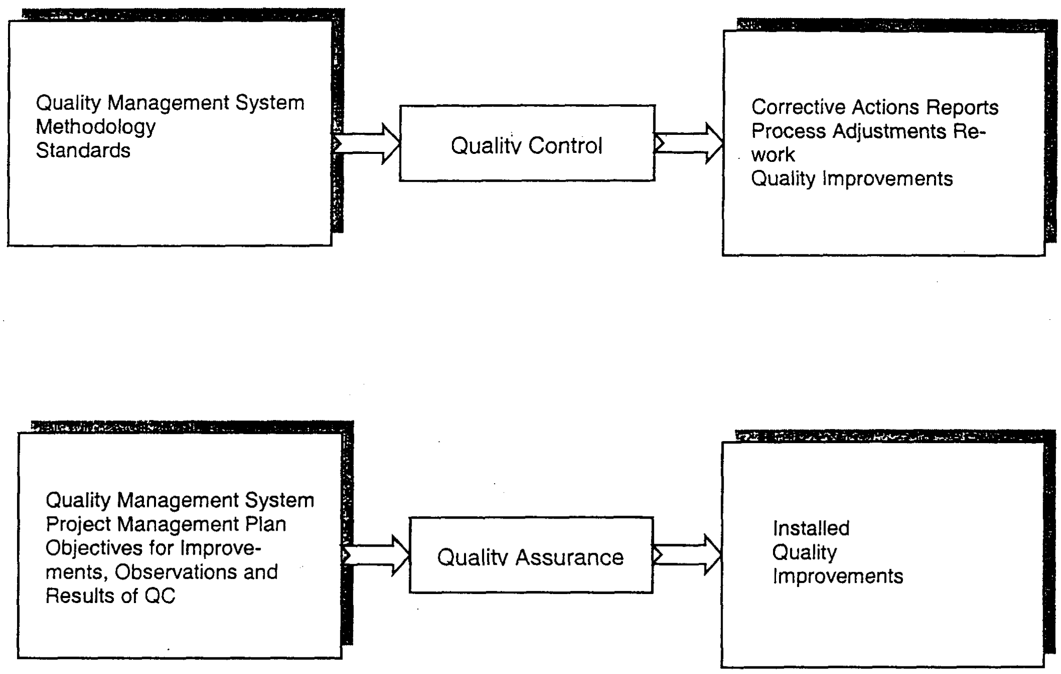
1. Quality Assurance Organization
2. Quality Assurance Plan
3. Procurement Control
4. Document Control
5. Control of Purchased Material, Equipment, and Services
6. Identification, Control, and Traceability of Materials, Parts, and Components
7. Control of Special Processes
8. Inspection
9. Handling and Storage
10. Inspection, Test, and Operating Status
11. Corrective Action
12. Quality Assurance Records
13. Audits

4.5.2 Quality Assurance and Control

Quality Control is the process used by the project team to meet the standards established by the organisation's quality policy. The process consist of observing performance, comparing it to the standard identified, and take the necessary action to correct the deviation observed. The cost of failing to prevent, detect, and subsequently correct deviations from the standard is known as the "cost of quality". It is widely acknowledged that keeping errors out of the process is far less expensive for the organisation than to deal with expensive process interruptions due to failure of keeping the prescribed criteria.

The basic tools used to control quality include: inspections, control charts, flowcharting and trend analysis.

Whereas quality control is concerned with monitoring specific results and eliminating causes of unsatisfactory performance, quality assurance is concerned with evaluation of the overall project performance to provide confidence that the project will satisfy the relevant quality standards. Therefore, to guarantee performance, the quality assurance process must address all the interfaces in the upstream operations or processes that are both internal and external to the organisation. This includes managing internal forces and external suppliers. The project team should document in the QA Manual the requirements against each supplier that will be evaluated. The team should also inform the suppliers about the process operations they will be involved in during the operation or process performance. In the figure below is the interrelationship of the inputs and outputs associated with QA and QC illustrated.



In **Annex 4.1** is given a practical example of a quality system for decommissioning – "Quality Assurance programme during the dismantling of the BR3 PWR reactor".

4.6 Human Resource Management

Project Human Resource Management is focused on all processes necessary to make the most effective use of the people involved in the project. In NPP decommissioning often the very best way in doing so, is to use the own NPP operational personnel to a maximum extent. We will discuss the organisational planning as determining and assigning project roles and responsibilities. Below, the organisational planning, the use of expert resources and the development of needed skills to enhance the project performance of the decommissioning project team, will be discussed.

4.6.1 Project Roles and Responsibilities

At the very earliest decommissioning project phase the project roles, responsibilities and closely related to them the reporting relationships should be developed. This organisational planning will be an impertinent part of the licensing application scope. In other words, the licensing authority will carefully check if the proposed project structure covers all necessary aspects of a safe decommissioning performance.

The organisational planning will be documented in the <u>Organisational Manual</u>. Throughout the project performance, the organisational structure must be regularly reviewed in order to ensure adequate applicability to the actual project phase.

4.6.2 Staffing the project team

Having worked out the organisation structure and the staffing requirements, the management team is responsible for getting those competent resources reliably assigned to and working for the project in the necessary time frame. It is a big difference in practice to know when and what resources will be needed, and to implement those in accordance with the project's needs and to achieve practical results.

Normally, one will take advantage of the existing plant operational organisation by using as much as possible the in-house staff. Part of the operational skills can be directly shifted into the decommissioning project's needs (radiation measurement, protection and control, post operation, waste management operation, decontamination, etc.). However, several specific tasks will have to be contracted externally (procurement of special tools and facilities, building demolition, special calculation within safety assessments, etc.). The project procurement management will be detailed in chapter 4.9.

4.6.3 Training and skills development

One of the very first actions of the project team will be focused on the qualification of the project organisation. Several questions immediately come to mind: Which associates are available to perform needed functions of project management? Do those available possess the right skills and experience? What training will be required? Will externals need to be hired? To what extent will external contractors be employed?

The initial approach should involve a review of available candidates within the company who are performing a project management-like function or have demonstrated multidisciplinary orientation. While the number of those candidates may be limited, there are advantages associated with the utilisation of own resources. They are already familiar with the company structure, processes, and clients. They should be able to assemble a management team quickly because they have already an understanding of the various departments. The moral of the employees will not be as negatively impacted as it might be, should external contractor resources be acquired.

4.7 Information Management System

This chapter deals with the importance of software routines to facilitate decommissioning project management processes. For presentation purposes the software systems developed and implemented to assist the project management of the KGR decommissioning project and describes the Project Management System (PMS) in detail as a central component of the developed software system.

Basic considerations

Before considering the project management process automation, the objectives of the system must be defined. Important part of this phrase is the word 'automation', which means "to function without human intervention". In this connection, the question immediately arises whether project management is at all possible without human intervention and if it is, to what extent it is useful.

In case automation is understood as the complete replacement of project management by the use of software, it is an unrealistic idea since the human ability to think critically, to respond, negotiate or to solve problems is indispensable for a successful project management.

However, it is useful to automate individual tasks provided that they run according to certain rules. This mainly concerns tasks which have to be carried out recurrently and which are time-consuming.

Part processes can be connected to subsystems. By a connection we understand the communication of part processes into one or both directions, which are initiated manually or event-controlled. By a system we understand the entirety of the part processes and connections necessary for a certain task (business process).

There is a large variety of software tools on the market, which cover certain project management processes. These software tools can be divided into basic classes:

- Word processing software
- Spreadsheets
- Accounting software
- Scheduling and tracking software
- Charting software
- Software development tools
- CAD

- Multimedia software
- Communication software
- Specific waste/material tracking software.

Furthermore, operating systems, netware and database software, which form the basis for the classes above, have to be mentioned.

This list is only a selection of basic classes and could be supplemented.

Numerous software packages on the market combine these classes, but the real challenge is the control of the overall decommissioning process.

The bigger and more complex a project is, the higher are the requirements on project management and the more attention has to be given to the question of software use.

The following information gives an idea of the scope and complexity of the KGR decommissioning process:

- The planned time period for the decommissioning project is approx. 12 years.
- Presently, approx. 7 700 tasks (as the smallest planning unit of the project – e.g. dismantling of a reactor head) are planned which are interrelated to each other. Since some of the long-term tasks have not been planned in detail, it can be assumed that more than 9 000 tasks will have to be implemented at the end of the project.

- Over the whole project time, the average own manpower of approx. 400 man/year has to be coordinated only for the direct dismantling and material logistic (without external services).

The use of information technique to support the management process can be of great advantage, provided that appropriate organisational and methodical frame conditions are established. The main advantages are:

- Processes can be schematised. Schematised processes have the advantage that they are more stable and less susceptible to errors and can reduce false activities.
- Reduction of expenditures for information procurement and distribution, minimization of paperwork.
- The central selected filing of information makes a central data security possible
- Increased availability of information inside (either for each employee or for a certain circle) or outside (authorised expert, authorities, public) the company
- Simultaneous working of several persons at one and the same process is possible, but changes immediately influence the work of all participants
- Aggregation of data, i.e. data of one level automatically summarize the data of the next lower level, in case they depend from each other
- Versatile use of registered information, i.e. information once registered can be evaluated in different connections
- Complex graphical evaluation possibilities
- The communication between project members can be improved (e-mail, reports etc.), also between members of different projects, as well as outside the company

- Easier understanding and control of decisions (project transparency)
- Fast adaptation in case of project changes (changed numerical data can immediately be calculated)
- Consequences due to changes can immediately be recognised
- Early identification of problems is facilitated as well as decision finding (timely dependencies of processes, critical paths, potential time delays, free resources and resource bottlenecks)
- Possibility to automate information flows. It mainly concerns the automatic actual data acquisition, the permanent planned-actual comparison and in case of important deviations (effects on the critical path) an appropriate escalation management.

4. 8 Risk Management

The Risk management Process

As a matter of fact, it is generally agreed that risk management is an integrated part of the project management process. On the one hand, traditional project management involves deciding what should be intended to be done (what should be in the Project plan), and managing the process of trying to do that. On the other hand, an effective Management Process involves recognising managing and reacting healthily to the uncertainties that are embedded in that plan (so called Variability or Category 1 Risk) and the uncertainties that may generate a need to modify that plan (Impact risk or Category 2 Risk). Thus, it is important to recognise that any project plan never will be ready to cope with every eventuality that might occur during the project performance.
Project risk management is not about avoiding risk, but rather about managing it. A provocative, but good definition of Project Risk Management is as follows:

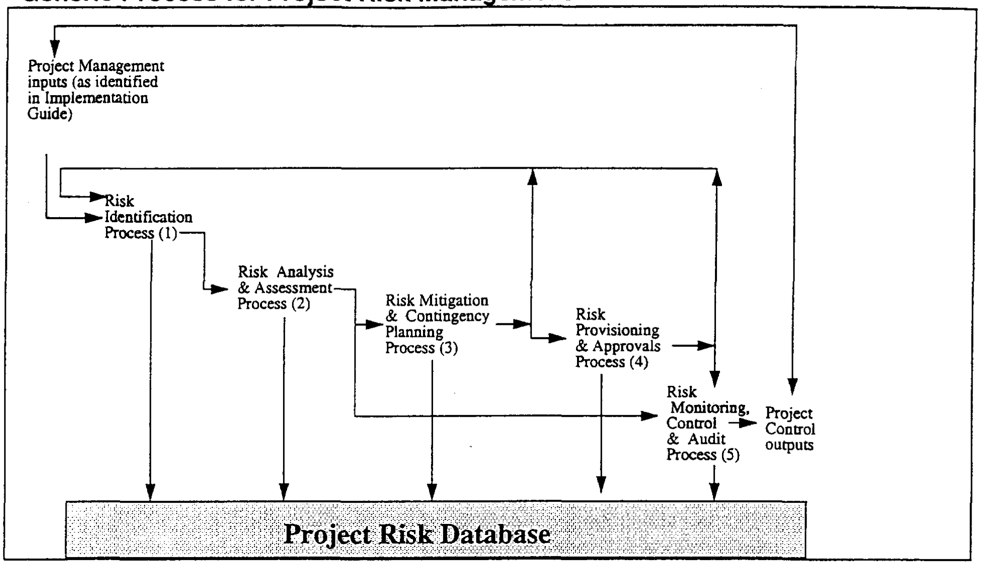
Project Risk Management (PRM) involves deliberately taking every risk that you should on terms that you can afford, and then proactively managing that risk to maximise stakeholders advantage [4.4].

Project Risk Management procedures have been published in various texts. As published it is usually straight-forward and involves a series of steps normally described as follows [4.2]:

- Risk Identification
- Risk assessment and Risk quantification
- Risk response planning (Mitigation and contingency Planning)
- Risk analyses
- Risk monitoring and control.

Risk Management is a process that commences with the identification of risks and links this through to the resolution of the individual risks. This process is shown in outline in the generic level drawing of the Risk Management Process, as illustrated below.

Generic Process for Project Risk Management



This process of project Risk Management operates within the project management process throughout the life cycle of the project. This process is complicated only by the interaction of the processes and not by the complications of the individual functions. Training of personnel in the risk management process is an important element of the risk management, as well as the design and utilisation of database(s).

4.9 Contract and Procurement Management

In planning the decommissioning project, services or goods will be identified which are not available or not possible to perform within the decommissioning organisation. These identified tasks might lead to procurement and contract activities. Depending from the cost value, there are several formal rules to be followed within the procurement process. Within the EC, above a certain value, the organisation is requested to perform a European tender process. If not being sufficiently familiar with such procedures, it could result in project delays and increasing costs. That alone would make it a very good reason for the management team to know procurement and contracting.

Within a decommissioning project, the team management leaders will undoubtedly become involved in carrying out some of these activities to customise, purchase, and secure equipment or professional services. Because any procurement project will have to be integrated into the overall decommissioning process and requires project team additional work (licensing, commission, control etc.), the team needs to acquire a basic understanding of the organisation's formal procurement management process, which establishes the framework and cross-functional foundation for the planning, scheduling and risk management of the procurement.

During decommissioning performance, equipment and/or services will be purchased/provided through competitive tendering. To ensure that the best product is bought at the best price, it is recommended that the project team follows the standardised procedures that have been established in their companies to select and manage external suppliers. Usually, within most companies, there is a specialised procurement department that leads the procurement process from the contracting point of view. Although project teams may become deeply involved in issuing and formatting the technical and preliminary contract documents, it is best to rely on the specialists from the procurement department and legal division to negotiate the final agreement. It is also recommended to integrate from the very beginning of the planning an expert from the licensing group to ensure consistency with the licensing frame conditions.

However, it must be noted that the quality of the services to be bought should always be given priority. Cheapest offer might lead to later delay in implementing the service due to cost consuming refitting works.

4.9.1 Procurement Management Process

Procurement is the process of acquiring needed services and/or products from outside the project's performing organisation. There are several process activities associated with project procurement management. These include:

- Procurement planning - determining what to procure and when
- Tendering planning - procurement documents and proposal's evaluation criteria
- Tendering – obtaining seller's response (bid, proposal)
- Source selection –prequalification procedures
- Contract administration – managing the relationship with the contractor
- Contract closeout – completion and settlement of the contract.

The procurement process could be formally divided into 3 phases:

(I) Planning phase (pre-acquisition)
(II) Execution phase (acquisition)
(III) Life-cycle phase (post acquisition).

(I) The planning phase contains 4 sequential but interrelated process steps:
Step 1 Definition of need.
Step 2 Specification of product or service to fulfil the defined need.
Step 3 Tendering and tendering response.
Step 4 Establishment of the contract with the vendor selected to satisfy the need.

4.9.2 Contract Administration

Contract administration is the process of ensuring that the contractors performance meets contractual requirements. In other words, contract administration includes all management activities necessary to integrate the contract performance process in the project overall management process.

4.9.2.1 The Contract

A contract is a mutually binding legal agreement to establish, cancel or change a legal relationship, which is achieved by concurring declaration of intention, i.e. application and acceptance, between two or more parties. It is distinguished between the seller (contractor), who is obligated to provide a specified product and the buyer, who is obligated to pay for this product. A contract includes all agreements made between the parties which are e.g. the proposal, the scope of work, the terms of payment, reporting, legal language etc.

4.9.2.2 Types of Contracts and Pricing Arrangements

Besides understanding the significance of the terms and conditions that appear in project contracts, it is equally important for project team leaders to understand how to choose a pricing arrangement that matches specific project situations. The type of contracts chosen for the project and its pricing arrangement depends primarily on the resources that the team has available to perform the tasks outlined in the statement of work and the risk associated with the work's completion.

The seller will want to make an as high a profit as the market will bear. The strategies and negotiation tactics need to align with the desired outcome and include the type of contract and pricing arrangements preferred by the buyer. For instance, to minimize the risk of agreeing to a higher-than-necessary price to cover the supplier's cost plus a reasonable profit, a risk-avoiding buyer will want to begin negotiating the cost of the work from the more optimistic range of possibilities. The risk-avoiding supplier, on the other hand, will want to avoid agreeing to a price that does not cover actual performance and push the price toward the more pessimistic range to cover all possible costs. Typically, actual costs exceed estimated costs, and neither party will want to assume the responsibility for the overruns. The seller's and buyer's determination to minimize this expense is ultimately an important consideration in the type of contract written.

Table 4.1 illustrates the relationship of the three fundamental types of contracts in the marketplace.

Table 4.1 TYPE OF CONTRACTS AND RISKS

Contract Type	High Risk to Buyer	Low Risk to Buyer	High Risk to Seller	Low Risk to Seller
Cost-reimbursement	x			x
Time and materials	x			x
Fixed-price		x	x	

Fixed-Price (FP) Contracts. Fixed-priced contracts are the simplest and most common form of standard business contracts that a project can risk on the supplier, buyers usually prefer them. Fixed-price contracts also offer the contracted seller greater opportunity to secure a substantial profit.

Cost Reimbursement (CR) Contracts. Cost reimbursement contracts are also known as cost-plus fee and require that the project leaders include in them (1) estimates of project cost, (2) provisions for reimbursing the supplier's expenses, and (3) provisions for paying a fee as profit. Cost reimbursement contracts are more favourable to the seller than to the buyer in that the buyer agrees to accept the risk of having to reimburse according to an acceptable allocation the seller's costs that may occur while the agreement is in force. This type of contract is used primarily in high technology R&D projects where there is high cost uncertainty, no definitive specifications, or the scope of work is vague. They are also preferred in the international market, where the risks of an unstable political environment make cost uncertain.

Time and Materials (T&M) and Unit Price Contracts. Time and materials contracts require project leaders to negotiate hourly rates for specified labour and to obtain agreement on the cost of all parts and materials. This type of contracts is frequently used to procure equipment maintenance and other support services, particularly when the time estimate to complete the set-up, repair, or overhaul of the equipment is uncertain. The buyer receives a bill for the number of labour hours at the agreed-on hourly rate and for the cost of materials and parts.

Be careful when negotiating T&M contracts because each hourly rate includes a component for overhead costs, which include both fixed and variable rate costs. Although variable rate costs can be calculated rather easily by estimating how many hours will be performed during the contract period, a share of the fixed costs, and the buyer benefits. Less frequently the hours are underestimated, and the supplier experiences a windfall profit. A technique that may be used to avoid this problem is to include in the contract a step-up discount plan that increases as the number of hours billed by the vendor increases.

A unit price contract is an arrangement by which the supplier is paid based on units of measurable output. A base floor and ceiling can be set and adjustments made to reflect price changes in the marketplace. These types of contracts are advantageous to both the seller and buyer because they are based on measurable costs; however, projects must establish fair rates and prices for the costs.

4.10 References

[4.1] IAEA Safety Guide No WS-G-2.1-4

[4.2] "A Guide to the Project Management Body of Knowledge", Newton Square, Pennsylvania USA, 2000 Edition

[4.3] OECD/IAEA/EC, "A Proposed Standardised List of Items for Costing Purposes", 1999

[4.4] B. Carter et al, "The European Project Risk Management Methodology", OPC 1996, Riskman

Annex 4.1

Quality Assurance programme during the dismantling of the BR3 PWR reactor

1. Introduction

1.1 Short history

During the preparation of the primary loop dismantling (the "Recovery loop") in 1997, it became clear that these operations would generate a very complex material flow, and that the management of this material flow would take an important place in the dismantling project.

In order to achieve an efficient management of the physic material as well as the relevant information of it, it was decided to set up a Quality Assurance programme.

As the dismantling and the materials management can be considered as a production process, the ISO 9002 was selected as QA system. At the beginning of 1997 the implementation of this QA system started.

1.2 The scope of the QA system

The project leaders preferred a step-to-step implementation and therefore it was decided to start with only the management of the material flow and the relevant information stream.

This resulted in the following scope:

"The Quality Assurance programme is applicable to the physical and administrative traceability of materials generated by the dismantling operations at the BR3 site. It concerns all materials of the controlled and uncontrolled area, with exception of the following ones:

- the effluents of the "gazeous waste";
- the effluents of the "Hotel Waste" (sanitary water);
- the daily waste from cleaning activities in the uncontrolled area, the metal and paper waste from cleaning the machines in the uncontrolled area included,
- the waste produced by administrative operations (mainly paper) from the different offices;
- organic waste from non sanitary activities (for instance clippings, grass...);
- high and middle activated waste from the dismantling of activated parts around the reactor core, the internal pieces and the Neutron Shield Tank;
- the irradiated fuel and fissile material;
- tools and equipment of external firms that are used and/or hired for specific operations."

Physical traceability means that for all material within the scope and produced during activities at the BR3 site, it is possible to know at any moment its origin, its location and which decontamination treatment it underwent.

Administrative traceability means that the information on activities and materials are documented, managed and that relevant information is communicated to the involved services.

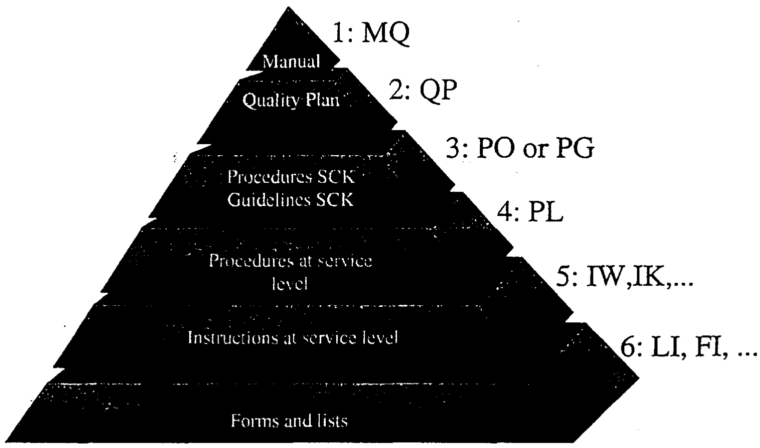
Although nuclear measurements are carried out on the dismantled material, the accuracy of the results is not (yet) part of the scope of this Quality Assurance programme. The management of the material flow of the dismantled material is, on the contrary, part of the scope.

2. The Quality Assurance programme at SCK•CEN

Since the beginning of the 90's and under the influence of some laboratories and services eager to obtain a quality certificate, the implementation of a general quality system at SCK•CEN was started. This QA system comprises two levels: the level "service" and the level "SCK•CEN".
The level "service" treats the implementation of the chosen QA system within this service. For the other quality requirements, not related to a specific service (i.e. training, purchase...) valid for different types of QA systems and being carried out by one specific service at SCK•CEN, a QA system on SCK•CEN level was implemented.

This twofold system leads to a strict hierarchy in the general SCK•CEN QA system, clearly shown in the structure of the document management system. The first three levels of documents are at SCK•CEN level and are valid for all services and laboratories of SCK•CEN; The other document levels are limited to the concerned service or laboratory.

The document management in this system consists of six levels.



Scheme of the QA system

• *Level 1: The quality manual - MQ*
This is a document describing in a general way the different elements of the QA and security system at SCK•CEN. It gives the mission and the lines of force of procedures and instructions.

• *Level 2: Quality plans - QP*
This is a document describing in a general way the different elements of which the system is build up in a specific service or project. It gives the lines of force of procedures and instructions as well as the supplements with respect to the Quality Manual. A quality plan can treat the quality, the safety as well as the environment.

• *Level 3: Organizational procedures or instructions – PO of PG*
These documents give more details about the organizational aspects of the quality and safety system (document management, treatment of complaints, organization for calibration, internal audit...) and define a series of acts or activities necessary to organize the system between different departments.
A procedure describes the "how", "the usual procedure", and sets thus the responsibilities and standards. Important is that a PO is valid for the whole SCK•CEN. Some procedures (PG) are valid for the whole SCK•CEN but are recommendations and they have no compulsory character as a PO.

• *Level 4: Procedures level "Service" - PL*
These procedures are complementary to the organizational procedures and are therefore more strict. They are valid for the complete service.

• *Level 5: Instructions - Ix*
These documents define how a more technical oriented activity has to be carried out. An instruction describes the "what" and is addressed to one or more persons or to specific machines or equipment.
Instructions are meant for persons executing specific tasks such as analyses or production steps (IW), calibrations (IK), maintenance works (IO), safety instructions (IV), environmental instructions (IE), specific research tasks (IR), instruction modules (IM)...

• *Level 6: Forms, lists and reference documents – FI, FR, LI, MT, REF*
These are described in the QA document, one level up. Reference documents (REF) and validation files (MT) are documents on which certain instructions are based.

3. The Quality System in sections D&D Dismantling and D&D Material management

3.1. Organization chart

The sections D&D Material management and Dismantling are two of the five sections of the Department Site Restoration, being itself part of the Radioactive Waste & Clean-up division.

To carry out the different dismantling tasks, the dismantling team per work is composed of personnel from both sections. This team is assisted by the different groups and subgroups of the sections.

3.2. Key persons in QA

Although all section members contribute to the good functioning of the quality assurance system in the section, the following persons have a key-role:

- The ***section heads*** have the final responsibility of the scientific output of the research domains and are also partly responsible to assure that the material leaving the site meets the requirements of the internal and external stakeholders. Moreover, in collaboration with the other staff members and during the weekly co-ordination meeting, they define the tasks to carry out and indicate the work responsible.
- The ***work responsible***, mostly a staff member, is responsible for all administration that has to be done before the proper dismantling, for composing the dismantling team, and for the co-ordination of this team during the execution of the works. The team members (operators, cleaning personnel, debtors) are chosen in the different subgroups.
  Already from the preparation works on, there is close contact between the work responsible and the Head of the group D&D Material follow-up because he is responsible for the material when the work is done. He is responsible for the management of the material flow as well as for the management of the information itself. Therefore he can ask assistance of different persons from different subgroups.
- The ***QA co-ordinator*** who assures that the QA system remains efficient.
- An (administrative) secretariat giving assistance for the practical side of all administrative formalities.

3.3. The customers or stakeholders of our QA system

Within our customers group, further on called stakeholders, a difference has to be made between those who need information in order to obtain and edit the necessary approvals and reports (the so-called information flow) and, on the other hand, those who will get the physical material (the so-called material flow) and this in function of the chosen evacuation way.

- **The stakeholders of the information flow**

For the sections D&D Material management and dismantling, the stakeholders are the Physical Control and the Technical liabilities (DRAB).

The physical control is responsible for the contacts with the authorities and the Approved Institutions, the nuclear safety, the waste characterization, the nuclear transports and the free release of material. In order to do this properly, both sections have to edit several documents giving information about the works and the produced material. These documents are:
- the work procedures;
- the qualification files for the waste (SL forms);

- free release files;
- files for fusion for free release or for reuse of dismantled material.

The Technical Liabilities (DRAB) is responsible for the contacts with NIRAS/ONDRAF and for the management of the waste within SCK•CEN. In order to do this properly, this service needs the necessary information from both sections to:

- keep the Belgian nuclear inventory up-to-date for the SCK•CEN part and more specific for BR3;
- efficiently manage the waste (radioactive and industrial) at SCK•CEN (using SL forms);
- arrange the transports to for instance the melting firms.

- **Stakeholders regarding the physical material**

Once the Physical Control and the Technical Liabilities have received the necessary approvals from the authorities and control organisms, the material can leave the site and be transported to its final destination. This can be:

- Belgoprocess, as subsidiary of NIRAS/ONDRAF, for the nuclear waste;
- the melting firms;
- the approved recycling firms (scrap dealers, chemical products, oils, bricks and debris...);
- the approved industrial dumps.

The Technical Liabilities (DRAB) is responsible for the organization of nuclear transports and transports to melting firms.
The transports to the scrap dealers is organized (after free release authorization of the Physical Control) by the SCK's Main Workshop (HWP) following the procedure "Management and recycling of reusable and unreusable materials, goods and scrap".

3.4. Quality in our organisation and working method

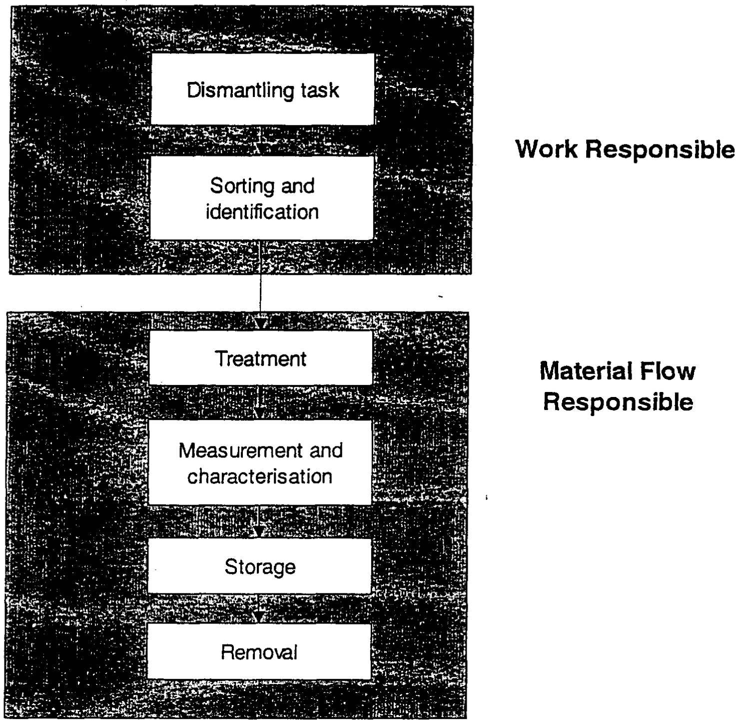
As already mentioned, the project leaders only wanted to have an ISO9002 quality assurance system for the traceability of the dismantled material and its relevant information.

Although our whole working methodology is described in several documents, there are three very important documents in our quality system.

- The first document is the quality manual of both sections. This document describes how the different ISO-requirements are met. It gives also the opportunity to refer to specific SCK-procedures for those requirements who are fulfilled by other services. Training, purchase internal audits, management of documents,... are typical such requirements.
- The second document is the service procedure (PL-document) that describes he practical execution of the operations and all the administrative work to do by the work responsible.
- The third important document is also the service procedure that describes the materials management. It describes how the physical traceability is maintained of the dis-

mantled materials during the different decontamination treatments and their movements on the site.

The chart below shows the whole process.



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- **The work part**

The weekly co-ordination meeting is an important tool in the quality system. During this meeting several specific quality actions can be discussed and proposed. Examples of such actions are an explanation of quality procedures, discussing of complaints, setting up of corrective actions...
Also in this meeting, the section heads Dismantling and D&D Material Management indicate mutually the work responsible.

This work responsible has to :

- write the work instruction, a document that describes the work and the team and that needs at least the approval from the Head of the Health Physics Service;
- write the accompanying report;
- arrange all other administrative formalities like entrance permissions;

- supervise the actual work; and
- transfer all the necessary data around the traceability of the batches to the Material Flow Responsible

Quality procedures describe the practical organisation of these type of works in the sections and also the structure and management of the different produced documents.

All documents related to the works as mentioned above are kept by the QA co-ordinator in a dismantling file. Such a dismantling file is related to mostly a loop or a part of the site (e.g. infrastructure) and contains all the available information about that loop. Examples of other information than the dismantling documents are the history of the circuit and contamination measurements.

- **Material follow-up**

The two main objectives of the material follow-up are to keep the physical and administrative traceability of the dismantled materials.

For keeping the physical traceability, a batch system was introduced. Each time when the work team creates a batch, this batch is foreseen of a sticker with an unique batch-number.
As long as the batch is awaiting its further treatment or further completion, this sticker stays on the batch.

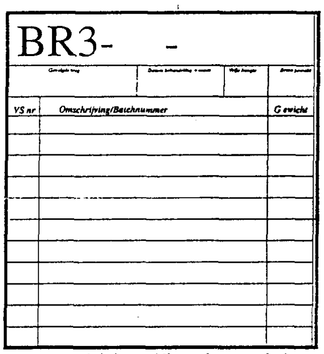

Figure 1: general identification sticker (sticker1)

When the batch is a drum, the following sticker is also placed on the side of the drum.

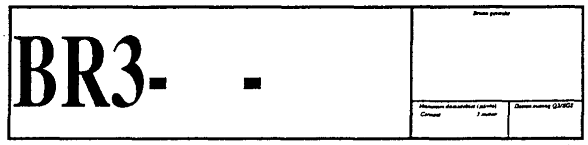

Figure 2: identification sticker for drums (sticker2)

The dismantling team and D&D Material follow-up fill in the different stickers with the following data:

- ***batch number***
- ***chosen evacuation way:*** radioactive waste; material for fusion; material for free release
- ***date of treatment and name of the person who carried it out***
- ***free height:*** (only for drums)
- ***VS-number:*** is the link between the work instruction and the dismantled material
- ***description of the material and weight:*** these two columns have to be completed accurately to guarantee a correct traceability
- ***gross weight:*** Once the batch completed, the gross weight is mentioned on the sticker.

The database (in which the batch number was already introduced) is now completed with all relevant information by the head of the group D&D Material follow-up, his assistant or a person indicated by the head of the group D&D Material follow-up. Each modification to the information (weight and/or composition) of a batch has to be mentioned in the logbook of the assistant Material follow-up.

For maintaining the administrative traceability all the necessary quality procedures and work instructions were written.
These comprises the management of the gained batch data with the database and the transfer of this data between several services of SCK (mainly the Technical Liability Service and the Health Physics Service). Also the local management of the final removal files (free release files, SL-forms for nuclear waste and melting files) requested by several SCK quality procedures is described in several procedures. These procedures stipulate exactly who fills in and manage the files, where the files are stored and how the information is given to the other services.

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Chapter 5

Financial Aspects of a decommissioning project

Chapter summary

Financial Aspects of a decommissioning project

5.1. Introduction

The objectives of a decommissioning cost estimate are to indicate the cost required to complete a decommissioning project, to determine the yearly budgets covering the expenditures and to allow further definition of the funding mechanisms. The decommissioning costs are estimated to make sure that sufficient funding is or will be available for the completion of the decommissioning project. The aim is to optimize the dismantling sequence and timing and thus to minimize the total decommissioning cost.

This chapter provides information about the component parts of a decommissioning cost estimate, discusses cost estimating methodologies and highlights the main factors influencing the cost estimate. The recommended process conducting to the cost estimate is also briefly described. An overview of financial mechanisms allowing to secure the decommissioning fund is also given.

5.2. Elements of decommissioning costs

5.2.1. Standard List of Items

Different costing methods have different data requirements, however, and consequently, their reliability depends on the extent to which various data are available and applicable to the specific case being considered. Independent of the assessment method, some uncertainty is inevitable in all estimates of future costs, and no costing method is generally superior to others in this respect.

Decommissioning cost estimates as published worldwide today show relatively large variations, by a factor of two or more for similar reactors. Even if it is generally recognised that the decommissioning costs can be estimated with reasonable accuracy, several international studies have been carried out to better understand the reasons for these variations.

Recently, the Nuclear Waste and Decommissioning Group of Experts of UNIPEDE has examined how different boundary conditions affect the costs of decommissioning [5.1]. To do that, different boundary conditions have been collected from twelve countries and costs for decommissioning a reference nuclear power plant have been estimated adopting the same methodology. Final results of the cost calculation vary almost by a factor of 6 from the lowest to the highest estimate, the most important factor of differences being the scope of the calculation. Also the decommissioning timing, the waste management system, administrative (labour rates and legal system) and financial (discount rate) factors have been recognised as important reasons of differences among decommissioning cost estimates.

To allow a more easy comparison between cost estimates, IAEA/EC/NEA proposed a "standardised list of items for costing purposes" **[5.2]** allowing to facilitate the comparison between decommissioning projects. The "standardised list" proposed eleven cost groups:

1. Pre-decommissioning actions
2. Facility shutdown activities
3. Procurement of general equipment and material
4. Dismantling activities
5. Waste treatment and disposal
6. Security, surveillance and maintenance
7. Site cleanup and landscaping
8. Project management, engineering and site support
9. Research and development
10. Fuel
11. Other costs

The list is then further broken down into tasks and subtasks. The detailed list of items is given in **[5.2]**.

It is commonly recognised that the costing of the groups 2, 4, 5 and 10 of the "standard list of items" requires setting up a detailed physical and radiological inventory under the form of a database. Typical relevant information is given in **Table 5.1** for a component (i.e. pipes, walls, etc.) that has to be dismantled. Attention has also to be paid to the inventory of the system fluids and the waste remaining from the exploitation period.

For a component to be dismantled, typical input data are:

- its localisation *i.e.* facility, building, zone, room;
- the average radiological conditions of the room concerned;
- its classification *i.e.* family and sub-family;
- its functionality by identification of the "system" or "function" to which it belongs;
- its physical characteristics *i.e.* material, weight, gross volume (when relevant), total surface, mean thickness;
- its radiological characteristics (this data are not always available prior to D&D; some of them also request sampling and analysing) *i.e.*
  - contamination type ($\beta\gamma$ or $\alpha\beta\gamma$) and level;
  - activation level;
  - typical dose rate;
  - identification of the isotopic vector describing the radiological inventory of the component.

Table 5.1: Typical form for inventory purpose of components to be dismantled

Inventory id				
Facility	Building	Zone	Room	
			Room Average Dose-rate (mSv/h)	Room Average Contamination (Bq/cm²)
Family	Sub Family	System		
Material	Mass (kg)	Gross Volume (m^3)	Total Surface (m^2)	Average Thickness (mm)
Contamination	Total Contaminated Surface (m^2)	Average Contaminated thickness (mm)	βγ Contamination level (Bq/cm²)	αβγ Contamination level (Bq/cm²)
OR				
Activation	Total activated volume (m^3)	Average activated thickness (mm)	Specific activation (Bq/g)	
Radiological stream: ref. Isotopic vector		Dose rate (mSv/h)		

5.2.2. Methodologies and techniques for decommissioning cost evaluation

In the past, the decommissioning cost estimates of nuclear power plants were related to the installed power. In fact, the decommissioning costs depend on the complexity of the nuclear installation, its physical and radiological inventory and local factors. One can identify 4 types of costs:

- costs based on detailed studies;
- work volume dependent costs;
- time dependent costs;
- specific costs.

Detailed studies are recommended for the cost evaluation of work packages requiring large engineering studies, preparation works and/or the use of special decommissioning techniques. This method should be applied for:

- the preparation of the installation before an important step in the decommissioning (e.g. preparation of safe storage period);
- the dismantling of the main components of the primary circuit of a reactor (*e.g.* reactor pressure vessel and reactor internals, main components of the primary circuit).

Work volume dependent costs are associated with dismantling, decontamination, waste packaging, transport and disposal of components. These costs can be easily estimated using the unit costs methodology. Unit costs or ratios are applied essentially to the mass or the surface of infrastructure, equipment or materials. It is for example the case for the evaluation of physical dismantling operations (ratios in euro/m, euro/m², euro/m³, euro/kg, etc.). To improve the accuracy of this method, a detailed list of unit costs can be set up to take into account categories and sub-categories to which the equipment and/of material belongs. A detailed inventory in the form of a database as explained in § 5.2.1 is necessary for this costing purpose.

Time dependent costs are associated with on-site management, administration, routine maintenance, safety, security, insurances, taxes and fees to Authorities and Regulatory Bodies.

Specific costs independent from quantities and duration are the costs for investments (*i.e.* decommissioning tools, dismantling and decontamination workshops, radiological survey, etc.), R&D, training of personnel, fees for the licensing process, contingencies, etc.

Generally a decommissioning cost estimate is a mix of the 4 cost types (first 4 bullets). The costs are evaluated in constant money (overnight costs). The advantage of having overnight costs and a standardised list of cost items is that comparisons and bench-mark are possible. The inflation rate will be accounted only afterwards, also for the funding mechanism definition.

Each cost is made up of 6 components *i.e.*:

- labour costs (including the workers salary, the allowances and the direct overhead costs);
- services (e.g. costs for work subcontracted outside the project);
- expenses (consumables for decommissioning tools, protective clothes, electricity, heating, telephone and water supplies, etc.);
- investment costs (under the form of a capital or capital rental or hiring cost in the case of unit costs methodology);
- secondary waste covering the waste produced by a decommissioning technique;
- contingency (cost to cover incomplete design or documentation, modification in the legal framework, changes in the market condition, tools/equipment breakdown, late delivery of supplies/equipment and adverse weather conditions).

The cost of a specific decommissioning task/component is given by:

$$\boxed{C_{\text{task/component}} = C_{\text{Labour}} + C_{\text{Services}} + C_{\text{Exepnses}} + C_{\text{Investment}} + C_{\text{Sec.Waste}} + C_{\text{Contingency}}}$$

with:

$$\boxed{C_{\text{Labour}} = T_N \,.\, \delta \,.\, HR_{OT} \,.\, (1 + OC) / \eta}$$

where:

- T_N is the duration of the intervention in normal conditions;
- δ is a coefficient (generally >1) accounting for the difficulty of the intervention (e.g. work with specific protective clothes, use of ladder or scaffolding);
- $HR_{\delta T}$ is the Hourly Rate of the specific Intervention Team (IT);
- OC is the Overhead Cost;
- η is the ratio between the effective working hours and the total working hours. This ratio accounts for the time required to meet the formalities for entering/leaving the controlled area.

$$\boxed{\begin{array}{ll} C_{\text{Services}} = T_N \,.\, \delta \,.\, HR_S & \text{(in case of time dependent cost)} \\ \textit{or} & \\ C_{\text{Services}} = Q \,.\, UC_Q & \text{(in case of volume dependent cost)} \end{array}}$$

where:

- HR_S is the Hourly Rate of the specific service;
- Q is the quantity;
- UC_Q is the unit cost for a specific task.

The other costs are given by:

$$\boxed{C_i = Q \,.\, UC_i}$$

where:

- UC_i is the "all-in" unit cost for expenses, capital rental or hiring cost or secondary waste.

This formula can also be applied for the primary waste management costs.

Also for the Contingency cost a formula can be provided. The formula is suggested by the AIF/NESP-036 guidelines **[5.8]** and is a percentage of the task cost (labour plus expenses costs) depending on the task/activity. An example is given in the following table:

Activity Category	Contingency, %
Engineering	15
Utility and DOC Costs	15
Decontamination	50
Contaminated Component Removal	25
Contaminated Concrete Removal	25
Steam Generator/Pressurizer/Circ. Pump Removal	25
Reactor Removal	75
Reactor Packaging	25
Reactor Shipping	25
Reactor Burial	50
Conventional Radwaste Packaging	10
Conventional Radwaste Shipping	15
Conventional Radwaste Burial	25
Clean Component Removal	15
Supplies/Consumables	25

The choice of a contingency rate depends on the knowledge of the nuclear facility, the decommissioning techniques and the expected evolution of the boundary conditions.

For the different costs, the most common estimating techniques are:

- the *bottom-up technique i.e.* the subdivision of the work in discrete tasks allowing to complete the decommissioning operation; for each discrete task a cost estimate is made;
- the *comparison technique i.e.* the deduction of the cost for the decommissioning of a specific component from the previous decommissioning of a similar component; if required, the cost can be adjusted to account for differences in radiological conditions, complexity and accessibility;
- the *parametric technique i.e.* the use of a model based on key driver parameters deduced from previous decommissioning experiences;
- the *expert opinion technique i.e.* several experts are iteratively consulted until a consensus cost estimate is reached.

5.2.3. Factors influencing the decommissioning cost estimate

Decommissioning cost estimates and funding mechanisms are influenced by political, social, economical and technical factors.

Political factors

The decommissioning activities have to meet the conventional and the nuclear safety requirements defined in the legislation. The safety requirements concern the protection of the workers, the public and the environment. More precisely the nuclear safety requirements are related to the annual maximal dose, the criteria for free release of material and the discharge of gaseous and liquid effluents. Changes in acceptable limits can have a significant impact on the cost estimates. So, a decrease of the free release level by a factor 10 generates roughly the following increases **[5.4]**:

- ± 5 % of the cost estimates;
- ± 17 % of the waste volume;
- ± 7 % of the doses.

The decommissioning of a nuclear installation may require the availability of 3 types of disposal facilities:

- a site for the burial of very low radioactive waste (see the French approach in absence of criteria for free release);
- a near-surface disposal facility for short lived nuclides bearing waste;
- a geological disposal facility for long lived waste and high level waste.

Unfortunately, in numerous countries, the disposal routes for decommissioning waste are not yet available and in some cases, even the availability of disposal facilities has not been (accurately) planned yet. Therefore, arrangements have to be made on the site or its vicinity to store the waste in a safe manner. The Regulatory Body or the Environmental Protection Agency can possibly require the conditioning of the waste prior to its future storage so as to reduce the risks of dispersion. Changes in the waste regulations may affect the cost of containers (for example different container qualifications) or the unit disposal fee (for example additional charges would be imposed by government for disposing off the waste in the repository).

The back-end solutions for the spent fuel and nuclear material can also severely disrupt the decommissioning operations and have an impact on the cost estimates. Indeed, some countries have decided to set a moratorium on the reprocessing of spent fuel and other nuclear material (*e.g.* plutonium). Spent fuel and nuclear material are then put into storage conditions in appropriate nuclear facilities, where they await their disposal after conditioning and repackaging, if required. The international practice recommends to evaluate separately the cost of back end solutions as in many cases also separate funds are set up.

Social factors

A significant factor influencing the cost estimate is the local social and economical condition. These conditions can play an important part in the decision-making concerning the degree of outsourcing, the use of new or high technologies and the choice of an immediate or a deferred dismantling strategy.

The involvement of stakeholders, from the public to the Authorities and Regulatory Bodies is rising from year to year. Information sessions are generally organised over the duration of the decommissioning project. These sessions have to be well prepared to avoid any disruption in the decommissioning process.

Economical factors

An aggregate cost estimate is composed by labour costs, service costs, expenses, investment costs, expenses, waste costs and contingencies. Each cost component has its own inflation rate over the decommissioning period. In many countries, the rise of the yearly waste cost significantly exceeds the inflation rate of labour and consumption products **[5.5]**. An average increase rate of more than 15 % is reported for the waste **[5.4]**. Such an evolution influences the cost breakdown of a decommissioning project. For example, if the cost estimates for the Biblis reactor made in 1977 and in 1991 are compared **[5.6]**, one observes that:

- the decommissioning cost estimate increases in a similar way as the inflation;
- in 1991 the waste cost represented 18,5 % of the aggregate cost instead of only 6,5 % in 1977;
- in 1991 the management (including licensing) cost represented 33,5 % of the aggregate cost instead of only 6,5 % in 1977;
- a decrease of the decommissioning costs by a factor 2.

Even if in 1977 the decommissioning experience was scarce, the general trends concerning waste and management costs observed in this study are confirmed in the last decade.

These economical factors have to be carefully evaluated, certainly in the case of a deferred decommissioning strategy.

Technical factors

The physical and radiological inventory of the installation can be considered as the main instrument for estimating the decommissioning cost. The size of the systems, structures and buildings will be a determinant factor for the importance of the D&D activities to be performed. The knowledge of the installation at the moment of the final shutdown is very important. A detailed analysis of the records, *i.e.* as built drawings, technical documents and exploitation records, gives an indication of the completeness and the accuracy of the inventory. Another aspect to be considered is the overall state of the infrastructure and equipment. It is important that it is noticed if parts of the infrastructure and/or equipment need to be refurbished or if their capacity needs to be improved taking into account the planning of the considered strategy.

The Waste Acceptance Criteria and waste management costs will influence the techniques to be used in order to optimize the decommissioning costs. For example, the size of the waste acceptable for disposal may allow less or impose more cutting operations for large components. High disposal costs may justify more decontamination efforts in order to free release materials (when such practice is authorized).

Nuclear sites generally contain several nuclear installations. Some of them can be of the same type, *e.g.* Pressurised Water Reactor (PWR) or Gas Cooled Reactor (GCR) etc. If their decommissioning can be planned in such a way that the same staff and the same decommissioning equipment and infrastructure can be used only with some adaptations, then costs can be saved on project management, training, licensing procedures and investments. The following ratios are used by EDF **[5.7]**:

- 5 % for the dismantling work cost;
- 2 % for the demolition work cost;
- 10 % for other works;
- x euro/file, set price for the adaptation of regulatory documents and depending on the type of facilities concerned.

The increase of the performances of decommissioning techniques is also something normal to be expected. This evolution can favourably influence the cost estimate or at least compensate for the evolution of the legislation over the decommissioning period.

5.2.4. Scheduling of the decommissioning costs

For the decommissioning strategy to be analysed, the corresponding detailed planning (see § 7.3) can be drawn. Then, the yearly budgets over the decommissioning project have to be calculated to cover the annual expenditures. These yearly budgets will then be used as input data to secure the decommissioning fund according to the adopted financing scheme (see § 7.3).

The funding of a decommissioning project and the decommissioning itself are both heavy and stretched over a long period. During this period, the costs of labour, waste, investments and expenses will increase according to their own specific escalation rates. Values of typical escalation rates can be found in specialised newspapers or reports from the National Institute for Statistics and from representative organizations or industries. This aspect has to be taken into account to determine the yearly budgets and the target value of the fund to be secured to cover the decommissioning costs.

5.3. Cost estimating guidelines

The first step in preparing a decommissioning cost estimate is to gather all the as-built drawings, construction records and exploitation records. This information allows to set-up a detailed physical inventory of the materials to be decommissioned. This inventory can then be completed by a radiological inventory based on the recorded incidental and/or accidental events, the activation in and around the reactor and a mapping of the contamination.

The second step concerns the definition of the boundary conditions of the decommissioning project (e.g. free release limits and discharge criteria for gazeous and liquid effluents, final objectives of the decommissioning project (e.g. green field or restricted reuse), availability of disposal sites for the various categories of produced waste).

The third step is to select the main decommissioning strategy and to define the waste management strategy allowing to optimise the decommissioning project in terms of radiological and industrial safety and in costs.

The fourth step (but mainly applicable when making a final cost estimate, i.e. just before starting the operations) consists in the set-up of a planning detailed into work packages, tasks and subtasks and taking into account the standardised list of items for costing purposes and the components recorded in the physical and radiological inventory.

This process gives as result the overall project duration, the critical path and the yearly detailed budgets in constant money value.

5.4. Financing mechanisms

The current practice worldwide is that the liability for the decommissioning of a facility and the site release remains on the Licensee. Similarly, as far as the waste management is concerned the "polluter pays" principle is usually applied. This principle requires that the costs of managing radioactive waste, from initial generation through to final disposal, are born by the organisations whose activities produce it.

Starting from this common principle there is however, as a result of particular circumstances in specific countries, a considerable divergence in the financing schemes adopted by different countries.

In the broadest sense, all the financing schemes applied to cover decommissioning and/or waste management expenditures, fall into one of the following categories:

- those that rely on a levy on electricity generation;
- those that rely on funds of the Licensee;
- those that rely on State or Government funding (historical legacy).

Within each of these broad categories, however, each individual scheme has unique characteristics. For example, Spain and Sweden both obtain the required funds by means of a levy on generated electricity but the collecting methods differ substantially. Thus, in Spain the whole of the electricity sector is levied as against only the nuclear sector in Sweden. Furthermore, different kinds of funds can be established, for instance for decommissioning, waste disposal and fuel back end.

The fund can be secured:

- before the start-up of the nuclear installation;
- by yearly provisions over the lifetime of the installation;
- by yearly budgets covering the decommissioning expenditures;
- by yearly payment of an insurance fee.

In any case, the financing scheme must ensure that sufficient money will always be available when it is needed without transferring an undue burden to future generations.

Whatever the financing scheme is, it is important to ensure a regular reassessment of costs. In fact, the cost estimates are usually permanently revaluated to be adapted to the latest data available. In such a way the assets and the projected decommissioning costs are permanently adjusted (generally on a yearly basis). The company responsible for the plant operation usually establishes the cost estimates but they are reviewed and agreed independently by Regulators or Ministries.

The resulting funds are either provisioned in the plant operator company accounts or in specific secured bank accounts or managed by an organisation in charge of radioactive waste management or by specific government funds. In this case, the national organisation also endorses the responsibility of the decommissioning.

The funds can be invested to earn interest, but in a secure way (against securities) so as to make sure that they will be available in due time.

Discount rate techniques are sometimes used to decrease the present liability. In this case, the overnight costs for each subtask or task using the techniques described in § 7.2.2 have to be inflated taking into account the specific escalation rate of the costs components and the year when the costs will occur. In the European community, the used discount rate techniques differ from country to country and within the same country even from company to company.

Table 5.3 gives an overview of the present situation in European countries.

More details can be found in reference **[5.7]**.

Table 5.3: Overview of the present situation in some European countries regarding the management of decommissioning fund

Country	Managed by	Constituted by	Discount rate	Revision
BELGIUM				
Utilities	Licensee owner	Yearly provisions	[1]	5 y
R&D centre	Licensee owner	Initial provision	0 %	0,5 y
Historical Liability	National Agency	Yearly expenditure budget	8 %	5 y
FINLAND	National Agency	Yearly provisions	0 %	1 y
FRANCE				
Utilities	Licensee owner	Initial provision	0 %	1 y
R&D centre	Licensee owner	Initial provision	-	-
GERMANY	Licensee owner	Yearly provisions	5,5 %	1 y
ITALY	State Owned Company	Levy on electricity price	-	-
SPAIN	State Owned Company	Levy on electricity price	2,5 % (real)	1 y
SWEDEN	State Owned Company	Yearly provisions	4 % until 2020 2,5 % after 2020	1 y

5.5. References

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[5.3.] IAEA, "Planning and Management for the Decommissioning of Research Reactors and Other Small Nuclear Facilities", Technical report series n°351, IAEA, Vienna, 1993.

[5.4.] Havard P., "Reduction of radioactive waste production: where is the optimum?", ICEM'01, Sept.30-Oct. 4, 2001, Bruges, Belgium.

[5.5.] J. Adler, P. Petrasch, "Decommissioning costs of light water nuclear power plants in Germany from 1977 to date", NIS, Contract FI2D-0051, Final report, EUR-14798-EN.

[5.6.] IAEA, "Operating experience and lessons learned on planning, organizational and management aspects of decommissioning", TECDOC in preparation, Vienna, 2002.

[1] Use of a capital rate

[5.7.] EDF, FRAMATOME, SCK•CEN and VUJE, "Study on the development of methodology for cost calculations and financial planning of decommissioning operations", study performed in the framework of a contract with the EC, November 2001.

[5.8.] AIF/NESP-036 Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates (prepared for the National Environmental Studies Project of the Atomic Industrial Forum by TLG Engineering) May 1986

EUNDETRAF

Chapter 6

Material Management

Chapter summary

Material Management

6.1 Basic conditions

6.1.1 Plant characteristics

In February 2001 the journal 'Nuclear Engineering International' reported that there were currently an estimated 439 reactors in operation generating 350 GW of electrical energy throughout the world. By 1990 a total of 48 power producing reactors had been shut down rising to 55 in 1992. To date, 90 commercial power reactors and over 250 research reactors have been retired from operation. This is set to steadily increase in number as nuclear plants come to the end of their useful operating life.
By itself the decommissioning of these reactors and those already shut down is an enormous task involving large amounts of manpower and money. However, the power reactors form just a fraction of the nuclear industry and the remaining facilities, most supporting the power reactors require consideration for decommissioning.
This includes facilities under the following types:

- Power reactors
- Research, prototype and Demonstration reactors built to support the nuclear power programme for individual countries
- Post irradiation and examination plants
- Nuclear fuel manufacture plants
- Reprocessing and storage plants
- Military plants supporting a country's military programs.

The list is by no means exhaustive but includes the major components of the nuclear power cycle and where most of the waste and materials will arise from decommissioning programs.

Nevertheless, the emphasis of the following is put to the dismantling of power reactors.

6.1.2 Definition of material quantities

The volumes of materials from dismantling vary with the size and the type of the reactor. To compare between different reactors a similar power output is used. However, this provides only general comparisons. The size of the core determines the types and amount of radioactive material generated. This is complicated by the position and design of the coolant systems used to take heat from the core and convert it into electricity.
In general, between reactors of similar power the Magnox reactor design will generate the most radioactive waste. Then in decreasing order of volume of waste comes the Advanced Gas-cooled reactor design (AGR), the water reactors (PWR and BWR) and with the smallest core is the Fast reactor. Influencing this sequence are the types of moderators, coolant and fuel enrichment which determine the power density of a core.
Magnox reactors have a low or nil U-235 enrichment. The fuel is metal uranium cooled by CO_2 and a graphite moderator. AGRs are enriched to about 3% U-235 and the fuel is in oxide form allowing it to operate at much higher temperatures.
Water reactor types are normally enriched to 3% or more and mostly moderated by light water, H_20 and in a few cases by heavy water, D_20. A liquid coolant allows high operating temperatures and fuel rods to be spaced close together.

Finally, Fast reactors require no moderator, have highly enriched fuel and a liquid metal coolant allowing a very high power density and hence small core.
The total number of operating hours and average neutron flux concentration in a reactor will generally determine the amounts in each category of waste according to the specific activity content. The longer the time spent in the core the more activation will occur. Similarly a higher neutron flux means increased bombardment of the core materials leading to higher activation levels.

The contamination in a plant which is indeed a spreading of activated particles will rise in the first years of plant operation, but will reach a nearly constant level after about ten years of operation.

The expected volume of radioactive waste is in general determined by
- type and quality of construction materials
- quality of the used nuclear fuel
- age of the plant
- operational history
- availability of clearance levels for materials.

Country	**Canada**	**Germany**		**Sweden**		**United States**	
Size and type of reactor	4 x 515 MWe PHWR	1200 MWe PWR	800 MWe BWR	900 MWe PWR	1OOOMWe BWR	1000 MWe PWR	1000 MWe BWR
Waste from 25-year operations*	6 900-27 500	6100-11000	6000-20000	6300	7500	21700	40000
Decommissioning wastes	10 000	6900	12400	700	15000	15200	16300
Total wastes* (operations and decommissioning)	16 900-37 500	13000-17900	18400-32400	13000	22500	36900	56300
Decommissioning wastes as a fraction of total waste	0.3-0.6	0.4-0.5	0.4 – 0.7	0.5	0.7	0.4	0.3

* Ranges in some estimates indicate the conceivable effect of possible incineration and compaction treatments.

Table 6.1: Estimates on the volumes of low and intermediate level waste from reactor operations and decommissioning in m³

Ref.: OECD Nuclear Energy Agency Report "Decommissioning of Nuclear Facilities: Feasibility, Needs and Costs", Paris, 1986

6.1.3 General conditions

Planning a decommissioning strategy has to take into account the general conditions of a plant. This means the possibilities of the plant itself in case of dismantling, material treatment, conditioning measures and storage capacities.

The distribution of internal and external work must be defined clearly. This often depends on the plant characteristics, but also on economic strategies. Provided that the financial opportunities are given, the main problem will be the logistics of material flow

within a nuclear plant, which was constructed for operation and not for dismantling. Obviously, a Boiling Water Reactor offers much more space for dismantling as a Pressurised Water Reactor, because the controlled area of the turbine hall can be used as a hot work shop.
But in any case dismantling starts with the removal of components to create more working space for further action.

The situation is also very different concerning the opportunities of depository and final storage. If nothing is available, the storage of material must be foreseen, e.g. on the plant site.

The decision for an internal or external dismantling and treatment of material will also depend on the quantity and quality of the staff which is available when the project starts. All situations are conceivable depending on the history of the plants post operation period.

Over all, the above parameters influence the strategy for decommissioning taken by the owner of nuclear facility. Strategies between countries differ because of several factors. National laws, political considerations and even population density all affect, to a degree, the strategy chosen by any particular utility.
Strategies can differ within countries depending on the types of facilities and primary requirements of the nuclear utilities. In addition, different nuclear facilities require different strategies. For instance the strategy for the decommissioning of a power reactor may be very different from the strategy for a fuel manufacture plant. Major considerations in any strategy are the type and amount of radioactivity and the safety requirements for the facility.

The discussion above highlights the complex nature in determining the materials arising from decommissioning. In general, utilities determine the decommissioning strategies for single facilities or a group of similar plants such as light water reactors. The large diversity of plants and their designs and usage necessitates that a general approach is adopted to identify materials.

6.2 Material characterisation

Wastes and materials from dismantling are controlled by several parameters that can alter the waste type and amount that is generated. The following paragraphs briefly discuss these parameters.

Primary wastes:

materials used to construct the nuclear facility. The total amount generated during decommissioning equals the amount used for construction but the amounts in the radioactive categories will vary according to the specific activity of the radio nuclides present in the waste. The effect of the dismantling technique must also be considered on the primary waste because of changes in activity content caused by cross contamination of the materials i.e. the spread and dilution of activity.

Secondary wastes:

This waste stream is heavily influenced by the dismantling technique. Waste categories are determined by specific activities which generally come from contamination of the secondary materials by the primary materials in the facility. Secondary wastes include the materials used to dismantle the facility such as handling/cutting tools, waste sentencing and disposal/storage routes, dismantling machines and support facilities, ventilation and filtration systems.

Whereas neutron induced activity is significant only where fuel has been present e.g. reactors, contamination is generally present in all nuclear facilities during operation. Contamination affects the categories of the primary dismantling waste generally by the amount of specific activity present. This in turn affects the dismantling technique considered appropriate and the subsequent handling, conditioning and packaging of the waste produced. Ultimately this determines the amount of secondary waste produced.

Timing of the dismantling operation can be significantly controlled by the principals of radioactive decay of the major nuclides present. In some cases, particularly for reactors, delaying decommissioning for a significant period of time after the fuel is removed will allow hands on dismantling techniques to be used rather than more expensive remote techniques.
The other consideration is that radioactive decay will affect the category of waste. This has major costs and handling implications for disposing or storing the waste. Waste with lower levels of activity generally tend to be cheaper to dispose of than waste with higher activity. In addition materials could be classed for re-use thereby minimising disposal costs further.
But on the other hand, if the time between shut-down and dismantling is too long, the γ-emitting and easy measurable Co-60 is not longer the main nuclide. In this case, there is more effort for radiation measurement e.g. for the declaration of activity or, more important, the control of activity incorporation of the personal during dismantling work.

6.3 Physical characterisation

6.3.1 Concrete

Concrete forms the largest proportion of the material arising from dismantling a reactor. In general, the specific activity of the concrete is low leading to most of it being disposed of as either general rubble or low specific activity waste.
Concrete is used to form the massive bioshield and, in the case of some AGRs and Magnox designs it also forms the containment vessel. Experience has shown that the concrete surrounding the reactor core is activated by the neutron flux. This activation generally only penetrates to less than 1 m in depth but can be complicated by the activation of any steel reinforcing material.
The Generation of radioactive Tritium by neutron activation of the impurity Lithium in concrete forms a significant part of the overall activity of the concrete. Tritium is highly mobile and can diffuse through the concrete. This leads to a spread of activity in the concrete and influences the volumes of concrete in each radioactive waste category and disposal of non radioactive material.

6.3.2 Steel

Steel forms the main structural components of a core and can be found in significant quantities within all reactors and in several forms according to the content of important elements such as nickel, carbon and chromium among others. In general steel is used

to generate the structural strength of the reactor and in some cases is used for the fuel cans.
Impurity levels within steel and major constituents of steel such as nickel can significantly affect the ultimate specific activity of the material after irradiation. Co-60, Fe-55, Ni-63 and Ni-59 are formed by neutron irradiation of stable elements and give rise to high radiation fields for several decades.

6.3.3 Graphite

Graphite is used as a moderator for gas cooled reactors such as the Magnox and AGR designs. The size of these reactors in comparison with similar power PWRs mean that the amounts of graphite for disposal are significant. The graphite can become activated by several routes, some depend on the levels of impurity. Consequently tritium is generated from impurities of Lithium present and C-14 is generated from the presence of C-13 and nitrogen, all by neutron activation.

6.3.4 Water

Water in two forms is used as a moderator and/or coolant by several rector designs. Both forms of water (H_20 and D_20) are in themselves fairly inert to neutron activation. However the use of a liquid coolant allows several nuclide types to dissolve into the water from the corrosion of the reactor structure. Hence Co-60, Cs-137, Fe-55 and Ni-63 are the main sources of radiation fields, the first two are the major sources.

6.3.5 Carbon dioxide

Carbon dioxide gas is used as a coolant. Like water, CO_2 is fairly inert to activation by neutrons apart from the C-13 $\rightarrow(\beta,\gamma)$ C-14 reaction which form a small amount of radioactivity in the gas. The main activity comes from particulate captured by the coolant circuit from the reactor structure similar in nature to that generated in water cooled reactors.

6.4 Radiological characterisation

6.4.1 General conditions

A reliable knowledge of the type and quantity of radionuclides together with their distribution and physical and chemical states is needed to define:
- the radiological protection of workers, the general public and the environment
- the operational techniques to be employed i.e. classification of operations as hands-on, semi-remote or fully remote
- the waste processing in terms of waste quantities, types in various national categories (activity bands), the treatment and conditioning processes and transportation requirements to meet national and international guidance
- the cost of the above.

In particular, since waste management usually dominates the overall cost of a decommissioning project, it is most important not to overcategorise wastes since this will result in excessive costs.
A combination of the above factors will enable the planner to define the strategy for decommissioning i. e. early decommissioning to IAEA Stage 3 or deferred dismantling.

A comprehensive characterisation process has the following steps:
- initial review of existing radiological information which exists as part of the control of reactor operations
- estimation of radiological inventory by calculation (often confined to activation products only)
- preparation of a sampling plan and subsequently the removal of materials in selected regions followed by their analyses for radionuclides
- carrying out direct measurements by radiation detectors and indirect measurements for contamination field assessment
- comparison of the measured data with those obtained by calculation
- feedback and iteration of the results to modify the calculations, adjust the sampling etc. until the appropriate quality criteria are met.

The results of the characterisation process provides the essential data set for assessment of the radiological hazards associated with decommissioning. Initially, the hazards associated with the characterisation process itself must be assessed - this is particularly relevant where basic data is not readily available from radiological surveys carried out during operations e.g. for removal of highly activated samples from a reactor core region. In such cases, scoping calculations should be carried out to assess the hazards present and the safeguards to be employed.

Once the data set for characterisation has been developed, further planning of the decommissioning work can be made to:
- provide dose and hence risk assessments
- assess the proposed operational methods for compliance with the ALARA principle
- identify the safeguards including the Personal Protective Equipment necessary to protect the decommissioning workers
- identify the safeguards required for the general public and the environment.

On the basis of the results of characterisation, the facility's structures and equipment can be classified according to radiation and contamination zones and forward work planned accordingly. The initial survey work is usually best performed in co-operation with operational staff who are familiar with the plant and understand its history e.g. the presence of contamination spills and clean-up operations which may have only been partially effective.

The process whereby radioactive nuclides are produced through irradiation by neutrons is termed neutron activation and occurs in materials located in or near the core of the reactor. The most highly activated materials are located directly in the core and lesser ones in the periphery e.g. the biological shield. Depending on the design of the reactor, some regions which are relatively remote from the core may become activated by the 'streaming' of neutrons through void regions e.g. in the gas ducts of gas-cooled reactors of the Magnox type. Neutron activated components constitute the greatest contribution to the overall inventory of the reactor amounting to tens of thousands of TBq for a commercial power reactor.

Generally, the most significant contributor to the gamma field in the early years following shutdown is Co-60 produced from the activation of Co-59. The presence of this nuclide largely controls the handling techniques which can be employed for decommissioning e.g. hands-on, semi-remote of fully remote operations.

Many other nuclides are also significant from the point of view of handling, transport and disposal (see Table 6.2).
Contamination is deposited on the internal and external surfaces of plant by the transportation of radionuclides produced by the erosion and corrosion of activated materials carried by the cooling circuits and/or fission products and actinides liberated from burst fuel rods. Contamination is of two general types:
- loose contamination capable of being removed by simple mechanical means such as wiping or scrubbing the surface or
- fixed contamination which is more tightly bonded to the surface and requires either removal by chemical decontamination or energetic mechanical methods such as jet blasting.

Loose contamination is generally the result of spillages from loss of coolant, decontamination processes or airborne releases. Other than in the gases of concrete and graphite where surface porosity occurs and some bulk removal may be necessary, generally, simple conventional cleaning techniques may be employed.

Parent	Nuclear reaction	Daughter nuclide	Principal Emissions	Half-life of daughter	Abundance of parent nuclide in patent element
				years	%
Li-6	n, α	H-3	β^-	12.3	7.5
C-13	n, γ	C-14	β^-	5730	1.1
N-14	n, p	C-14	β^-	5730	99.6
Na-23	n, 2n	Na-22	β^+, EC	2.6	100
Na-23	γ, n	Na-22	β^+, EC	2.6	100
Cl-35	n, γ	Cl-36	β^- (β^+, EC)	301000	75.8
K-39	n, p	Ar-39	β^-	269	93.3
Ca-40	n, γ	Ca-41	EC	103000	96.9
Fe-54	n, p	Fe-55	EC, γ	2.7	5.9
Mn-55	n, 2n	Mn-54	EC, γ	0.86	100
Fe-54	n, γ	Fe-55	EC, X	2.7	5.9
Ni-58	n, γ	Ni-59	EC, X	76000	68.3
Ni-62	n, γ	Ni-63	β^-	100	3.6
Co-59	n, γ	Co-60	β^-γ	5.3	100
Zn-64	n, γ	Zn-65	EC, β^+	0.67	48.6
Zr-92	n, γ	Zr-93	β^-	1500000	17.1
Mo-92	n, γ	Mo-93	EC, X	3500	14.8
Nb-93	n, γ	Nb-93m	IT, X	16.1	100
Nb-93	n, γ	Nb-94	β^-, γ	20000	100
Mo-94	n, p	Nb-94	β^-, γ	20000	9.3
Mo-98	n, γ	Tc-99	β^-	213000	24.1
Ag-107	n, γ	Ag-108m	EC, γ	127	51.8
Ag-109	n, γ	Ag-110m	β^-, γ	0.68	48.2
Sn-124	n, γ	Sb-125	β^-, γ	2.76	5.8
Ba-132	n, γ	Ba-133	EC, X, γ	10.5	0.1
Eu-151	n, γ	Eu-152	EC, X, β^-, γ	13.3	47.8
Eu-153	n, γ	Eu-154	β^-, γ, X	8.8	52.2
Eu-154	n, γ	Eu-155	β^-, γ, X	4.76	0
Ho-165	n, γ	Ho-166m	β^-, γ, X	1200	100

Table 6.2: Significant Activation Nuclides for Decommissioning

Fixed contamination on the internal surfaces of pipework becomes a problem for removal in reactors - particularly in direct cycle plants such as BWR's and RBMK reactors where 'crud' deposits of ferritic oxides build up through erosion and corrosion of the primary circuit. The nature of such materials is dependent on the reactor type and removal of this contamination inventory, which can amount to tens or hundreds of TBq, is an important early step in the decommissioning process. At stagnation points in the coolant flow or points of pressure drop caused by discontinuities in the flow, crud deposits tend to accumulate. In addition to the activation products which are deposited in the crud, fission products and actinides may also be present where fuel has failed during reactor operations. The quantities of these species are dependent on the dwell time before removal of the failed fuel or, if the reactor has experienced accidents during its operations, significant quantities of fission products and actinides will be present. Major fission products and actinides are presented in Table 6.3.

Radionuclide	Half-life	Principal decay
Sr-90	28.5 y	Via Y-90, β^-
Tc-99	2.1E+05 y	β^-
Ru-106	374 d	β^-
I-129	1.6E+07 y	β^-
Cs-137	30 y	β^-
Ce-144	285 d	β^-
Pu-239, Pu-239, Pu240	87.7 a, 24110 y, 14.3 a	α
Am-241	432 y	α
Cm-242, Cm-244	162.8 d, 18.1 a	α
U-232, U-236, U-238	69 y, 2.3 E+07 y, 4.4E+09 y	α

Table 6.3: Major fission products and Actinides

Time from shutdown, and the obvious effect that this has on the quantities of the shorter lived radionuclides, has an important bearing on the overall inventory.
For a reactor at shutdown, the major activation products in steels are Fe-55, Co-60, Ni-59, Ni-63 and Nb-94, for reinforced concrete; H-3, C-14, Ca-41, Fe-55, Co-60, Eu-152 and Eu-154, for graphite; H-3, C-14, Eu-152 and Eu-154.
Some typical inventories of power reactors of various standard types are given in Table 6.4.

Type	Location	Radioactive inventory (Bq)	Assumptions
BWR	Caorso, Italy	1.8E+16	2590 MW_{el}, 7 years irradiation 4 EFPY
PWR	Trino, Italy	5.7E+15	870 MW_{el}, 23 years irradiation, 10.6 EFPY, 5 years after shutdown
Magnox GCR	Berkeley, UK	5.4E+16	355 MW_{el}, 26 years irradiation, 5 years after shutdown
UNGG GCR	St. Laurent-2, France	1.4E+16	1830 MW,,, 21 years irradiation, 14 EFPY, 6 years after shutdown
VVER 440	Greifswald-1, Germany	1.3E+16	1350 MW_{el}, 17 years irradiation, 9 EFPY (approx), 6 years after shutdown
CANDU	Canada	6.3E+14	98.5 MW_{el}, 21 years irradiation, 17.1 EFPY, 5 years after shutdown

EFPY: Effective Full Power Years

Table 6.4: Typical Inventories of Power Reactors for Major Components

Co-60 is the predominant gamma radiation hazard and this nuclide together with Fe-55 accounts for most of the inventory (Bq) in the first 10 years after shutdown. After 50 years, the longer lived isotopes of nickel, niobium and silver dominate. In concrete and graphite, tritium dominates in the short term leaving C-14, Ca-41 and europium isotopes to dominate in the longer term. Contamination fields in the short and medium term are generally dominated by H-3, Fe-55, Ni-63, Co-60, Sr-90 and Cs-137. Actinides will also be present and become more significant in the longer term. In general, the mix of nuclides and quantities may vary over a wide range depending on:

- reactor type, power history and shutdown period
- composition of construction materials and particularly the presence of trace elements
- operational methods e.g. decontamination, corrosion rates influenced by coolant chemistry
- unplanned events e.g. fuel failures, spills etc.

6.4.2 Use of Computer Codes to Calculate Activation Inventories

Calculations of neutron induced activities are used routinely for the initial estimation of reactor inventories. Such calculations require an estimation of the spatial and energy distribution of neutrons which flow through the materials. These neutron fluxes are then used to determine the reaction rates of the parent elements which generate the various radionuclides. The reaction rates are used to calculate specific activities of the various nuclides based on the composition of materials, power history and decay times. The data requirements for the process are summarised below:

- power/time histograms defining the reactor power history
- cross-section data for neutron energies and temperatures
- fuel characteristics - geometry, enrichment, burnup
- dimensions, masses and materials compositions of reactor components subjected to the neutron flux
- Decay period following shutdown.

The computer codes utilised can be classified according to those which estimate the spatial and energy distribution of neutrons and those which take this information and use it to calculate the induced activities. The literature on these codes is extensive and the reader is referred to specialised publications for further information e.g. the codes, ANISN, DOT/DORT and MCBEND are all popularly used for the calculation of neutron energies and distributions. Similarly, codes such as ORIGEN2 are used to calculate the induced activity in system materials. Once the induced activity has been calculated this can then be used to calculate the dose rate from components such as MICROSHIELD or RANKERN.

Uncertainties in the calculations when compared to the results of measurements arise for a variety of reasons:

- inadequacies in the methodologies employed e.g. use of a single dimension model when two or three dimensional models would be more appropriate
- modelling approximations e.g. control rod positions and histories
- geometry simplification, differences in design and as-built geometries
- neutron source uncertainties
- neutron streaming effects in cavities
- inaccuracies in the application of cross-section data
- inaccuracies in materials composition data e.g. trace impurities in construction data including inhomogeneities in samples removed.

In general, good agreement can be achieved in the fuelled region of the reactor core (to a factor of two) and agreement gets progressively worse away from the core in the axial and radial directions.

6.4.3 In-situ Measurements

Measurements can take the form of either direct measurements of radiation, contamination measurements or spectrometric measurements for individual or several radionuclides. In the latter two cases, the measurements can be carried out directly or on samples removed for measurement. The type of method employed depends on:

- the type and quantity of the radiation emitted
- the physical and geometric conditions
- whether quantitative or qualitative measurements are required.

A summary of data needed, the use it is normally put to and methods of collection are presented in Table 6.5.

Data needs	Specific uses of data	Data collection methods
Radiation (a, ß, y) dose or exposure rates	Necessary to identify radiation hazards and access limitations, to specify decommissioning procedures and methods, and to estimate waste volumes	Direct radiation measurements, screening level, air monitoring
Amount of loose and fixed and fixed contamination an surfaces	Necessary to evaluate effectiveness of pre-decontamination, to plan protection against airborne releases and to identify personnel protection measures	Analyses of smear samples and correlated . radiation measurements
Location of radiation sources and contamination (`hot spots')	Necessary to evaluate design sequence of decommissioning actions, to specify decommissioning procedures and methods	Direct radiation scans, historic knowledge of process
Contaminant penetration into walls and floors	Necessary to design sequence of decommissioning actions, to specify decommissioning procedures and methods	Scans and analyses of core samples
Contamination levels in soils under and near the facility	Necessary to specify decommissioning procedures and methods, to assess foundation removal and excavation hazards	Analyses of soil samples historical soil sampling data

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Table 6.5: Data Needs, Uses and Collection Methods for Inventory Information

6.4.4 Dose rate measurements

Dose rate measurements can be used to establish the activity of a waste item if important calibration factors are established e.g.

- Predetermined distance from source to detector
- Fixed isotopic composition and distribution of activity
- Fixed surface geometry
- Identical detector set up and sensitivity parameters
- Fixed background noise.

Such measurements are often termed `fingerprinting' whereby once the waste parameters and detector set up is fixed then a range of waste items can be surveyed and the results extrapolated on a 'like for like' basis.

6.4.5 Contamination measurements

Contamination is either loose or fixed depending on its adherent properties. Loose contamination may be measured by rubbing the surface with a filter paper (usually 100 or 300 cm') and transferring this to a counting device. Fixed contamination can be measured by placing a fixed detector at a given separation from the surface and monitoring the radiation reading. Alternatively, for large contaminated areas then scanning devices may be used. In this latter case, the detector integration time must be matched to the scan rate to prevent spurious readings (typically< 5 cm/s).

6.4.6 Spectrometric measurements

Spectrometric measurements are routinely used in the nuclear field to determine isotopic composition. Small portable instruments may be used in the field for gamma measurements. Generally, alpha and Beta spectrometric measurements are confined to the laboratory. Using a similar set of calibrations to those required for the dose rate measurements described above, it is possible to determine quantitatively those gamma isotopes present which are important to the decommissioning process by reference to calibration standards. Often the gamma spectrum is used to determine the presence of beta emitting isotopes through scale factors determined via sampling and analyses. The easiest measurements to perform by far are those for gamma emitting isotopes using standard equipment as follows:

- sodium iodide detectors (NaI(Tl) - for applications were low resolution gamma spectrum information is required
- germanium (Ge(Li)) or intrinsic silicon detectors for high resolution spectra.

Further descriptions of equipment are not discussed here and the reader is referred to the specialist literature and manufacturers' specifications.

6.4.7 Sampling and Analysis

Analytical work is often carried out either to cross check the accuracy of calculations or direct measurements. Additionally, alpha and beta emitting radionuclides are not readily determined by direct measurements and samples have to be removed and transferred to a radiochemical laboratory for analysis. Certain important radionuclides for decommissioning are hard to measure and cannot be determined other than by radiochemical separation and specialist measurement techniques. This is particularly so for electron capture nuclides such as Fe-55 and Ni-59 and beta emitters such as H-3 and C-14 where these radionuclides occur in combination with others. In summary, the main requirements for sampling and analyses are:
- verification of theoretical calculations
- estimation of surface contamination fields
- development of correlation factors (`fingerprints') for hard-to-detect nuclides.

Sampling is expensive since it usually requires specialist tools, often built on a 'one-off basis which require development and testing before deployment. Samples invariably require special packaging and shielding in approved containers before they can be transported to the analytical facilities. All these requirements contribute to the cost of the work. Additionally, it is usually difficult to determine the number and positioning of samples to enable a statistically meaningful result to be obtained. Hence a sampling programme is needed to define the samples required e.g. `biased' sampling may be used to adequately characterise a region where known activation or contamination hotspots occur. Alternatively, `unbiased' sampling can be used where the general effects are thought to be homogeneous.
A cost effective approach determined by the required quality of the results should be applied pragmatically. For unknown contamination distributions an iterative approach has to be adapted with the sampling adapted according to the results obtained.

Analyses require access to specialist radiochemical facilities which provide the capability to carry out chemical separations followed by alpha, beta and gamma spectrometry. The know-how to select the appropriate techniques is essential, hence highly qualified staff are normally required for other than routine measurements which can be carried out by technicians.

6.4.8 Example: BR 3

As an example the following describes the radiological characteristics of material arising from decommissioning the Belgian Pressurized Water Reactor BR3.

Two aspects are important during decommissioning of a nuclear reactor. The first one is the dose rate aspect which in the case of a PWR is dominated by the γ-radiation from Co-60. The second one is the contamination aspect for which not only the γ emitting nuclides such as Cs-137 and Co-60 are important but also the presence of α contamination can present a particular issue.
For waste management, it is also important to determine the so-called critical nuclides i.e. the nuclides which are difficult to measure (the pure β nuclides such as Ni-63 and Ni-59, Sr-90, Nb-94, C-14, H-3... and the α nuclides such as the Am-241 and the Pu and U isotopes) and which are an issue during long term disposal due to their long lives and their specific radio-toxicity.
The determination of these critical nuclides in waste packages is a difficult task. Several approaches are followed to satisfy the disposal requirements. Estimations on the basis of neutron activation calculations, materials composition and irradiation history allow to determine rather precisely the activation levels for the major components of the irradiated materials such as the Ni and Fe isotopes for metals and Ca isotopes for the concrete. For elements present at trace levels such as Nb, C, Tritium, Eu..., the accuracy of the estimation depends strongly on the exact original content of these trace elements which is generally not known precisely.

Radiochemical measurements are the best way to determine the exact radiochemical composition of the activated materials. For BR3, during the dismantling of the high active internals, samples were taken systematically during the cutting operations. Some swarf material was collected during the cutting operation and subjected to detailed radiochemical characterisation. The radiochemical determination implies a complex analytical work with a series of separations to eliminate the strong γ nuclides which are present in activity levels several orders of magnitudes higher than the investigated isotopes.

A still more difficult task is to estimate the critical nuclides coming from fuel leakages which are fission products such as Sr-90 and Cs-137 or all the α isotopes. This is quite impossible to model so that only radiochemical determinations can solve the problem. This requires the estimation of a mean surface contamination level in α, β, γ-emitters and the determination of specific contamination isotopes such as the Cs-137 γ emitter, the correlated Sr-90 β emitter, the determination of the α spectroscopic composition including the Am-241, the long lived Pu, U and Cm isotopes as well as the β Pu-241 emitter.
For BR3, the composition of two mean surface contamination levels were determined:
- the first one, the high contaminated level composition, is representative of the primary pieces which were never decontaminated;

- the second one is representative of the pieces which were decontaminated during the Full System Decontamination of the primary loop which was the first step in the dismantling strategy.

Table 6.6 gives an overview of the radiochemical isotope vectors which were derived for different waste streams. (Reference date: 1998-07-01 i.e. 11 years after shutdown)

Correlation	Thermal shield	"Vulcain" internals	"Westing-house" internals	Contamination vector
Ni-63/Co-60	1.4E+00	3.6E+00	7.1E+00	1.1E+00
Ni-59/Ni-63	2.8E-03	1.9E-03	3.6E-02	2.0E-03
Fe-55/Co-60	1.2E+00	3.8E+00	4.8E-02	1.8E+00
Nb-94/Co-60	4.4E-05	1.3E-04	2.3E-04	4.0E-03
C-14/Co-60	1.1E-04	4.3E-04	7.2E-04	4.2E-03
H-3/Co-60	1.3E-04	5.2E-05	3.2E-04	3.2E-04
Cl-36/Co-60	4.6E-06	1.3E-05	2.7E-05	3.7E-06
Sb-125/Co-60				1.9E-03
Tc-99/Co-60				5.9E-06
Sr-90/Cs-137				3.7E+04
Am-241/α_{tot}				4.6E-01
Pu-238/α_{tot}				3.5E-01
Pu-239+240/α_{tot}				1.5E-01
Pu-240/Pu 239				3.8E+04
Pu-242/Pu 239				3.4E-03
Cm-244/α_{tot}				4.0E-02
Pu-241/Am 241				4.3E+01
U_{tot}/α_{tot}				4.0E-03

Table 6.6: Radio-nuclide vectors of different waste streams in BR 3

6.5 Material flow description

6.5.1 General conditions

The availability and nature of a country's waste management infrastructure and provisions are important factors for a decommissioning project. Some of these factors are:
- existence of a waste repository or at least
- defined acceptance criteria for radioactive waste
- criteria or clearance levels for unrestricted release of material.

The non-availability of waste disposal repositories capable of taking the quantity and types of radioactive wastes resulting from operation and decommissioning may preclude the option of applying an early dismantling strategy and may require the construction of waste storage facilities at the nuclear power plant site.

Where a suitable repository is available, the waste acceptance criteria applicable to that repository have an impact on decommissioning. For example, where a repository will accept the direct disposal of large components, and the transport infrastructure will allow this, removal of large components is an economic feature. Alliteratively, where components have to be reduced in size and placed into standard disposal packages, then space and features to accommodate the necessary size reduction and waste packaging equipment are necessary. In addition, different waste acceptance criteria may apply to different waste categories and this affects the effort required for the handling, size reduction, packaging and storage of these waste components.

Some countries allow or require the recycling of certain radioactive waste materials and set waste acceptance criteria for this. The recycling of lightly radioactive metals is an example of such a practice. This may involve the melting of the metals in furnaces and the size and capacity of available furnaces may have an effect on the degree of size reduction required in dismantling.

A similar example relates to the availability of clearance levels for solid materials. Clearance is an administrative procedure whereby waste below a certain level of concentration (Bq/g or Bq/cm^2) or amount is deemed to be below regulatory concern. The further use of such materials may be unrestricted or restricted to specific applications.
If waste clearance criteria have not been established, it may be necessary to classify as radioactive waste a greater proportion of the waste resulting from dismantling. The associated waste management and storage facilities required at the time of decommissioning may consequently need to be larger.
Efforts are under way in the IAEA to help reach international consensus on clearance levels for decommissioning waste and on monitoring programmes to verify compliance with these criteria. Guidance on various approaches regarding the unrestricted or restricted release of solid materials, including decommissioning waste, can be found in various IAEA publications.

6.5.2 Material flow: example KRB A

The dismantling work of contaminated and activated components in the German NPP KRB A will end up in about 10,000 tons of material. At the moment more than 7,100 tons are already dismantled and treated, about 54 % could be given to unrestricted release, 32 % were used for restricted recycling (melting) and only 14 % waste were generated, mostly the activated components like reactor pressure vessel and biological shield.

The contamination for water effected systems was up to 50,000 Bq/cm^2 and for steam effected systems about 1,000 Bq/cm^2. Approximately 60 % of the contaminated scrap could be cleaned for unrestricted reuse or free recycling.
The extremely low limits of the official German requirements require strong efforts with respect to decontamination and measurement techniques, and they represent the main factors influencing the costs. About 33 % of the steel parts with residual specific activities above 0.1 Bq/g - mainly pipes and valves with small diameters and materials with complicated geometry - were used for "nuclear recycling" by melting.

At a very early stage already one has to decide about the way of treatment of the dismantled components. Mainly with respect to costs all factors like conditioning and packaging as well as decontamination and documentation are taken into account to find the economic way for recycling or final storage. The total amount of only 14 % of waste of the total scrap shows the successful application of the developed and optimised decontamination techniques and measuring procedures (see figure 6.1).

Table 6.7 shows the situation for some representative systems. The savings are up to 70 % in case of recycling concrete and even in case of high contaminated material more than 30 % of costs can be saved.

water effected systems	steam effected systems	electric cables	concrete
70 %	50 %	55%	30 %

Table 6.7: Comparison of the effort for recycling and final disposal (Costs for final storage are excluded)

The actual experience of recycling 86 % of about 7100 tons of dismantled material in KRB A and the consequent control of costs gives us the reason to prefer the way of recycling for contaminated materials in most cases.

6.5.3 Waste minimisation

6.5.3.1 Depository and legal conditions

Radioactive wastes from decommissioning consist mainly of solid materials together with a small percentage of liquids. Solids comprise of materials such as pipes, armatures, mountings, tanks, other metallic components and supports, cables, insulation, worn-out and damaged equipment, concrete and rubble. Liquid materials mainly arise from decontamination work and from water purification systems. Conditioning of all of these wastes includes processing and/or packaging if necessary after sorting or pre-treatment. For this purpose, a variety of proven as well as advanced conditioning techniques are available.
In gėneral, materials arising in the course of decommissioning a nuclear facility will be treated the same way as materials arising during the operation of those facilities. The bulk mass can be regarded as non-active material. About 5 % of the weight is to be classified as active material.

Besides legal aspects, the management of materials arising during the decommissioning of nuclear facilities is an optimisation problem with cost for final or interim disposal on one hand and cost for achieving reuse or disposal as non-radioactive material on the other. Especially, costs for proving compliance with release criteria are to be considered, although some of these costs can be balanced against savings made in final disposal charges.
Re-use can for example be achieved by decontamination or by separating those parts of a component that cannot be decontaminated easily. If the site, of the nuclear facility to be decommissioned, is planned to be used further then consideration should be given to re-use of buildings. Re-use after some kind of treatment is also possible, i.e. electric cables or motors etc. Metallic material can be melted and subjected to further

production processes (recycling within the normal commercial markets without or alternatively under certain 'restrictions' within the nuclear field).

There are two main lines: disposal as non-radioactive material and disposal as radioactive material. For the first line compliance with legal requirements concerning the release of material has to be fulfilled, often resulting in extraordinary high efforts for measurements of activity, but on the other hand, resulting in a minimisation of radioactive wastes and thus saving volumes for interim and final disposal. Radioactive waste is to be disposed of in special disposal facilities that need, as the experiences show up to date, time-consuming and labour intensive licensing procedures. Sometimes final disposal facilities are near the surface, mostly they are situated in deep geological formations, so far most are still under planning or construction.

The design of disposal facilities is based on radiation protection features relating to short term aspects during the operation of such a facility as well as to long term aspects relating to the release of activity to the environment after the end of the operation period. These aspects lead to a complex system of requirements that have to be fulfilled by the waste that is to be disposed of. In addition, as far as a disposal site is not situated on the site of the nuclear facility requirements for the safe transportation of wastes have to be complied with.

6.5.3.2 Technical measures and methods

The process of conditioning of radioactive wastes aims at the production of such waste packages that comply with the above mentioned requirements. Most often, the waste itself does not fulfil all requirements and therefore requires being packed into waste containers. The unit from waste product and waste container is described in the following chapters as waste package.
Materials as they arise after segmentation out of plant systems etc. are called raw waste. In the course of treatment processes the raw waste is changed into waste products. Selecting the appropriate treatment line is facilitated by previous sorting into waste categories. These material categories show similar or identical behaviour in the applied physical, chemical or biological treatment processes. Sorting before treatment and packaging also avoids or at least lowers the occurrence of chemical reactions in the final waste product or within the final waste package.
The choice of an appropriate technique for the conditioning of a certain type of waste depends on the physical and chemical properties of the waste itself. The activity content as well as the nuclide composition are the key factors in designing the equipment for radiation protection related to minimising radiation fields in the vicinity of the process plant as well as possible releases to the environment. Other factors are related to space, time and man-power requirements.

Solid materials may be classified into metallic and non-metallic materials. They may arise as large single items or as bulk masses and may contain some moisture. Several treatment modes can be applied to dried solid wastes being based on two different treatment processes, namely compaction and incineration. Wet solid wastes normally need some pre- treatment in order to lower moisture content to acceptable limits.

Possible treatment modes are:
- No compaction, no incineration
- Pre-compaction, no incineration
- Pre-compaction and incineration
- Super-compaction and incineration incl. super-compaction of ashes
- No-compaction, incineration.

As can be seen, compaction and incineration may be complementary but the actual choice depends on technical and economical factors. Pre-compaction of combustible wastes prior to incineration leads to lower transport volumes, but higher efforts related to man-power and technical equipment because of the additional handling. Moreover, it may require additional treatment before incineration can be started. Otherwise burning of the materials will not be optimised. So, pre-compacted wastes may show less advantageous incineration features than loose or not compacted wastes since shredding of the pre-compacted wastes before incineration could become necessary.

Super-compaction is generally regarded as the more robust process in comparison to incineration that needs extraordinary efforts in the sorting of materials. The proportion of non combustible waste in one shipment is as a rule limited, often it shall not exceed a limit of about 10 to 20 % of the weight.
Moreover, sophisticated techniques are necessary in the case of incineration facilities to assure that stringent requirements concerning the release of activity and chemical compounds into the environment are met. Normally, free liquids within the waste are not allowed. Because incineration facilities are working within so-called campaigns and are delivered from different nuclear facilities, the question of cross-contamination arises and is of great importance when changing from the incineration of alpha-bearing wastes with high contents of transuranic elements to the incineration of typical wastes out of nuclear power plants. Incineration plants most often are designed for a certain range of waste material properties, latest developments tend to aim toward a wider range of materials. The main advantage of incineration is the elimination of all organic and combustible material from waste thus resulting in very high volume reduction of those wastes to be disposed of. Additionally, there is no biological activity left in the waste and the risk of fire during the storage period is avoided. To achieve further volume reduction residues from the incineration can be super-compacted. The decision on which method has to be applied for waste treatment is often derived from legal aspects and licensing conditions, such as limitations to occupational exposures.

Pure metallic materials such as black and stainless steel, aluminium, lead, copper, etc. can be melted and depending on their activity content, melting times and volume reduction with regard to final disposal volumes can be used in different kinds of recycling (within the nuclear or commercial markets). Therefore and to meet the requirements of the melting process itself segmenting to appropriate sizes and processes of sorting are necessary.

Another alternative may be seen in the application of fixation or encapsulation processes. The best time to perform such encapsulation may be a matter of debate. If it is carried out prior to the availability of disposal facilities, there is a risk that expensive re-treatment of the waste may be required once the disposal requirements are finally specified. Delaying treatment however, prolongs storage of raw wastes when they are better stored in inert and packaged forms. The level of confidence between the nuclear industry and the regulatory bodies in specifying packaging requirements in advance of selecting a disposal site is the key factor in deciding when to package raw wastes.

Liquid wastes, e.g. water arising during decontamination processes, coolants during segmentation work, oil, etc. are normally conditioned by transforming them into solid or nearly solid products. Two principles are decision-making for these processes:
- Due to the transformation into a (nearly) solid state the risk of spread and release of activity during the transportation and the storage period (leachability) are substantially reduced.
- The transformation into (nearly) solid products results most often into a significant volume reduction. In those cases where fixation e.g. cementation) is applied this effect may be partially neutralised or reversed.

The choice of an appropriate process may rely on the well-known liquid-solid separation processes as they are implemented throughout various industrial applications. Depending on the chemical and physical properties and the solid content in the liquid wastes one of the following processes or a combination of them may be selected:
- Incineration
- Fixation
- Filtration, Centrifugation, Decantation
- Evaporation.

One example for a combination of some of the above mentioned processes is the evaporation of inorganic liquids followed by drying of the evaporation concentrate resulting in solid waste with a moisture content of about 10% of the weight. Another is the evaporation of liquids followed by a process of fixation (cementation, bituminization).

The choice of a combination of processes may depend on the importance of the different aims as it is the case for solid materials too. For example, fixation with the help of special materials always results in higher volumes of the final product in comparison to the input waste material. The decision for such a volume augmenting process may be based on the costs directly related to the process itself (including the costs for transportation) and on the fact that there will not be any risk of being short of disposal facilities now and in the future. But, because disposal facilities cannot be regarded as so-called 'free goods', normally, it is necessary to apply volume reduction techniques and processes. This may be enforced by legal aspects and for radiation protection reasons (also with respect to a minimisation of transportation and related risks).

In some cases, fixation can result in an overall volume reduction. This is the case when different materials are joined at the end of the conditioning line. One example is the combined conditioning of powder-like resins and evaporation concentrates during the final process step of drying. Another example is the conditioning of waste building materials where contaminated waste water may be used in mixing waste material with fresh cement to allow an optimised filling of the final waste container.

Fixation is not clearly defined, it may rely on pouring techniques to fix solid materials within a waste package and may include processes such as cementation, bituminization, or embedding into an organic matrix. The latter ones are often named as encapsulation that is to say, embedding of small particles into an appropriate matrix, which aims for example on minimising solubility and leachability.

6.5.4 *Waste conditioning methods*

Decommissioning of nuclear facilities can be achieved in different ways and there may be different final status of decommissioned facilities. Application of conditioning techniques will be enforced at different stages, at different times and to different extents as a function of the selected decommissioning alternative.

The selection of conditioning techniques is based on an extensive data gathering in the nuclear facility to be decommissioned. This data would include data of components to be dismantled, types of materials (e.g. concrete, stainless steel, black steel, non-ferrous metals, plastics, etc.), activity contents and type of activity. Moreover, requirement from disposal and recycling concepts have to be taken into consideration, they may result in a constraint of the choice of packaging and segmenting procedures. It is clearly understood that common factors such as costs, man-power, waste quantities and radiation exposure of the personnel have to be considered.
In the planning of a decommissioning project several working steps have to be identified. The task of conditioning, transportation, and final disposition of radioactive wastes is one of the longest lasting steps in a decommissioning project.

Constraints and interdependencies to other decommissioning activities become obvious, for example with respect to the dismantling sequence. Variations adopted to the specific situation in respect to available techniques, disposal routes and legal requirements (working protection etc.) may present optimised solutions.

Material	Possible Treatment	Internal Packaging and Transportation	External Packaging and Transportation
mixed material, e.g cable, electric components, small parts	(super-) compaction	pellets	container for pellets (suitable for final disposal)
metallic components	measurements for unrestricted release	small containers (boxes)	Container
	decontamination		
	melting	200-1 drums	
	final disposal	small containers (boxes)	container for final disposal
isolation material	(super-) compaction	pellets	container for pellets (suitable for final disposal)
concrete: blocks, debris	decontamination	small container	container (suitable for final disposal)
	release		
	final disposal		
secondary wastes: protection clothes, foils, etc.	incineration or similar processes	bags	container for final disposal (after super-compaction of ashes)

Table 6.8: Typical but simplified overview on materials as they arise in the course of decommissioning and possible treatment modes

Single conditioning techniques should be implemented directly into the planning of a single segmentation task. Aspects of concern are summarised as follows:

- Sequence of working steps for dismantling, segmentation, and handling.
- Requirements and extraordinary measures that have to be complied with or that have to be taken.
- Selection of an appropriate waste container with respect to transportation requirements and final disposal.
- Evaluation of an optimum filling of containers and the necessary number of containers.
- Radiological check of the inventory of the containers.

Selection of the appropriate types of containers has to be done very timely because of the necessary manufacturing times.

The task of conditioning of radioactive wastes is closely connected to the different steps in the process of decommissioning and will be performed to a great extent in parallel with other tasks. Principle technical features and processes must be clarified before actual dismantling and decontamination work begins. Today, a wide range of such techniques exists.

6.5.4.1 Conditioning methods for solid materials

Compaction

Compaction processes are generally carried out with drum compactors that may have either horizontal or vertical pistons. Compactors may be stationary or mobile. If the process uses extremely high compaction forces to give large volume reduction factors the process is termed super-compaction.
The process can be summarised as follows: A steel drum filled with solid wastes is placed on the compactor platform and the compactor disc is hydraulically rammed into the drum (in-drum compaction) or, alternatively, may crush complete waste drums. In general, applied forces are in the range of 1000 to 2000 Mg. Related to the specific area of a drum's lid pressures are in the range of 10 to 100 N/mm². Special devices for in-drum compaction of ion exchange resins work at pressures of about 20 Mg.
The force required for super-compaction has been investigated through experiments where it was shown that for all types of waste composition nearly maximum compaction could be achieved. The required forces for super-compaction were also compared with the force that may be experienced by the package under disposal conditions in deep underground repositories. These geological forces were estimated to be approximately 300 bar for the KONRAD mine in Germany.
Compaction with lower force is normally aimed at minimising transport volumes or at maximising packing density within a waste container.

Supercompactor with horizontal piston

A supercompactor with horizontal piston can be easily transported in containers to the place where waste arises. The main parts are:
- ram with hydraulic device
- jib crane and grab

- pre-compactor
- exhaust device for charger and exit.

The height of the plant with jib crane is about 2.8 m, with the pre-compactor being about 4 m. The mass without the pre-compactor is about 47 Mg and with about 50 Mg.
The supercompactor is constructed from steel and is painted to aid decontamination. Beneath the cylinder is a device for collecting fluids that may flow out during compaction (see figure 6.2).
The working pressure is slightly above 300 bar. The maximum area for compaction is 1 m x 0,54 m. Larger diameter drums can be reduced to the correct diameter using the pre-compactor. The pre-compactor, with a mass of about 5 Mg, is positioned above the charge opening on the working platform. During compaction aerosols may be produced. These are removed by mobile filtering devices at the charging position, the exit and above the collecting device for liquids.
During compaction all data concerning the waste and the operational features are collated and recorded.
There are several possible modes of operation:
- Compaction of wastes delivered in 180-ltr or 200-ltr-drums
- Compaction of wastes contained in transport drums which are tipped out into the compactor and compacted into special cartridges
- Compaction of wastes that have been delivered in bags.

200-ltr-drums are pre-compacted whereas 180-ltr-drums can be directly super-compacted. The main advantage of this procedure is the enclosure of the whole activity during the process.

Loose debris is compacted by tipping the contents of the transport drum into the super-compaction channel using a hydraulic drum grabber. Before charging, a cartridge is put into the compaction channel via the slide and the lid of the cartridge is placed directly in front of the ram. The waste is then filled into the channel and compacted into the cartridge which is finally sealed with a lid. The final product is a steel mantled pellet.

Wastes delivered in bags are compacted as described above. All pellets are measured automatically. Collated data contains the mass and height of each pellet and the dose rate (measured manually). The pellets are then marked and packed into a drum or into a container with the help of a grabber or a crane. Following several campaigns, figures can be derived for the effective volume reduction factor. Depending on the type of waste mixture that is compacted, typical volume reductions are given in Table 6.9.

Nuclear power plant	**A**	**B**	**C**
Volume of raw waste	20 m³	15 m³	14 m³
mass (wet)	3310 kg	6500 kg	2510 kg
volume after compaction	4 m³	6.3 m³	3.2 m³
mass (dry)	3145 kg	6100 kg	2380 kg
Volume reduction	5	2.4	4.5

Table 6.9: Results from several campaigns of supercompaction

Typical throughputs are in the range of 10-30 180-ltr drums per shift (8 hours of operation per day). The normal operational staff consists of four workers.

Supercompactor with vertical piston

Wastes are filled into drums, transported to the working platform, surrounded by a pressure bell and compacted with a force of about 15,000 kN. The products are pellets with the same shape as described in the previous section. The pellets are transported to a short-time buffer awaiting final packaging into containers or 200-ltr-drums.

Low Force Compactors

As a pre-treatment step, burnable wastes are sometimes pre-compacted into bales prior to incineration. This method is generally used to reduce storage and transportation costs. However, recent incineration campaigns have shown that pre-compacted wastes burn less readily than loose wastes. A drawing of a vertical compactor for in-drum compaction is shown in figure 6.3.

Figure 6.4 shows the implementation of such a compactor within the conditioning line for spent resins as applied in a German nuclear power plant. The main characteristics of this process are summarised as follows:
The moisture content of the waste is about 50 % of the total weight. During the filling of the drum build-up of cavities occurs. In order to minimise the volume of these cavities a force of about 160 kN is applied. The process of filling and compacting is repeated until the desired filling level is reached. The maximum that one drum with a net volume of about 500-ltr. can contain is 180 kg of dry material.

Auxiliary equipment

As mentioned previously, free liquids may be forced out of the waste during compaction. Pellets (cartridges) containing waste that have discharged free liquids have to be dried in order to fulfil special requirements for interim or final storage. The plant for drying pellets is called PETRA (Pellet-Trocknungsanlage). Its main operational features are summarised below. Main Components:

- Drying and Heating Chamber
- Device for Operation at Low Pressure and Condensation Unit
- Energy Supply and Data Processing Unit
- Cooling Device
- Drum Weighing Device.

Up to eight drums can be placed into the heating chamber. Its dimensions are 3,4 m x 1,8 m x 2,5 m (length x width x height) . Electrical heating is performed by elements present in the back wall of the facility within an air channel. Movement of air is carried out by two ventilators. Maximum operating temperatures are up to 300°C. Insulation of the heating chamber is assured by a 80 mm thick layer of mineral wool. If necessary the outer surfaces of the casing and the main construction parts can be painted for easy decontamination. The heating chamber is constructed of galvanised carbon steel with a zinc layer.

Drying is carried out at low pressures of about 20-50 mbar absolute. Vaporised water is condensed in the condensation chamber with a total condensation surface of about 1,5 m^2 and finally released to the water collecting device approximately every ten minutes. The dimensions of the cooling device are 1,0 m x 1,0 m x 1,5 m. A glycol-water mixture at a temperature of about 5°C is used as coolant.
Before and after drying, drums are weighed with a drum scale. This and other process related data are buffered in a data processing unit, so that data for obtaining approval for interim or long-term storage can be made available.

Drying is stopped when the following criteria are fulfilled:
- inner pressure of drums: < 50 mbar
- temperature at drum outer surface > 130 °C
- amount of condensate per 200 1 drum < 100 ml/hr.

In order to ensure that drying is complete the drums are left for another two to three hours before being removed from the PETRA facility.
Whereas this system is used after super-compaction, drying systems may be implemented before super-compaction. The technical principles are very similar.

Incineration of solid wastes

Many techniques for the incineration of solid wastes are used throughout the world, e.g. shaft kiln furnaces with stationary or movable grates and rotary kilns. In Europe, mainly shaft kiln furnaces are used. Therefore, in the following sections short descriptions of shaft kiln furnaces are given, including details of sorting of the wastes before incineration and offgas cleaning.

Shaft Kiln with a tippable grate

Incineration in a Shaft Kiln with a tippable grate is carried out in the primary chamber and all ashes from the incineration are collected in the ash box (see figure 6.5).
The initial temperature of the furnace is 200 to 300°C and it is heated to a temperature of about 800°C. When the incinerator has reached 800°C the charging of burnable waste is started. Batches of about 20 kg are fed into the incinerator at intervals of about 5 - 10 minutes. Charging is carried out via a sluice and the off-gases of the sluice are fed back into the primary chamber. The waste starts to burn. After a few charges the furnace temperature exceeds 800°C and the primary oil burner is shut off. The temperature of the primary chamber is then kept constant at 800 – 1000°C by regulating the incineration air.
The off-gas leaves the primary chamber and enters the secondary chamber. In order to obtain a complete incineration, the off-gas passes the secondary oil burner, where air is added and a continuous flame is burning. The off-gas leaves the furnace and is cooled down to about 200°C by passing through a "water to gas" heat exchanger. The off-gas is then cleaned in the bag filter of solid particles. A large fraction of the small amount of activity contained in the off-gas is separated together with these particles.
Cleaning of conventionally hazardous compounds such as dioxin and SO_2 from the off-gas is carried out by the injection of lime and charcoal into the off-gas tube. Dust particles containing dioxin compounds etc. are filtered out in the second bag filter.

Before being released to the environment via the stack off-gases are continuously sampled and monitored.

In a single campaign about 5 - 6 Mg of waste can be incinerated producing 500 kg of ash and about 50 kg of dust from the first bag filter. The masses that arise from filtering in the second bag filter during cooling are negligible.

The majority of the activity is contained within the ashes and the dust filtered out in the first bag filter. The activity content of the waste delivered for incineration is very low, leading to a dose rate at the surface of the bags of the order of μSv/hr. Control of the activity flow and the activity content of the products is carried out by a gamma-scanning of the ash-drums, gamma, beta and alpha measurements of samples of the products in a laboratory and by continuous dose-rate measurements during operation.

After an operation period of several days, the ashes are removed from the primary chamber by hydraulic lowering and tipping of the bottom of the chamber. Ashes fall down into the ash collector and, after allowing time for cooling, are loaded into the ash-drums with the help of long-handled tools. Repeated analyses of ash samples have shown the remaining combustible material in the ashes to be approximately l% by weight.

Shaft Kiln with moveable grate

The incinerator with a shaft kiln with moveable grate consists of a decomposition chamber and a combustion chamber which are separated by a moveable grate (see figure 6.6). The plant does not work continuously. Feeding of the decomposition chamber and removal of ash from the combustion chamber is carried via transfer chambers.
Near the grate the temperature is about 800°C but in the upper parts of the furnace temperatures are slightly higher. Variation of the temperature can be achieved by adjusting the primary air inlet depending on the temperature at the grate.
Initial decomposition of the waste is achieved by adding very small amounts of air. This process is known as pyrolysis. The resulting gases are sucked into the combustion chamber and are burnt by an excess of secondary air which is fed in through the grate. Solid particles not yet decomposed and non-combustible parts of the waste material fall into the combustion chamber whose temperature is controlled by input of secondary air. Before filling up the decomposition chamber, the combustion chamber and the off-gas ducts are preheated to more than 800°C. Then waste is fed into the decomposition chamber and spontaneous ignition occurs. Three process stages may be identified: drying, degassing and gasifying. Products from this second decomposition contain high-molecular tars and oils that are cracked while passing over the hot bed. Following cracking, these gases consist mainly of CO, CO_2, H_2, H_20 and smaller amounts Of CH_4. The permitted level of CO and organic C in the off-gas cannot be exceeded.
Off-gases leaving the combustion chamber have a temperature of about 900°C and are cooled down in a heat-exchanger to 300°C. Afterwards they pass a prefilter. Gases leaving the prefilter show dust contents of lower than 100mg/m^3- Further cooling of the off-gas is reached by adding fresh air resulting in a temperature of about 100°C.
The last filtering device related to activity is represented by an aerosol filter (HEPA-filter). The total fraction of activity released to the environment in relation to the activity leaving the combustion chamber is about 10^{-5}.

In addition this incineration plant contains a scrubber to filter out halogens (chlorides, fluorides) out of the non-radioactive off-gas. Because of the high efficiency of the HEPA-filter, liquid residues from the scrubber can be released via the chemical clarification plant (neutralisation). Tritium is retained in the wash solution up to about 30%. The capacity of the plant is in the range of 70 - 100 kg solid waste per hour. It depends on the caloric value of the wastes normally being in the range of 4000 - 10000 kJ/kg. From the technical point of view it is possible to incinerate oil and solvents (up to 20 1/h) together with solid wastes. But, because of the close proximity of the incineration plant for liquid materials, whose off-gas treatment is specially designed for retaining halogens, no use is made from this option.

6.5.4.2 Pouring of solid materials

Solid materials that are not compactable and that require uneconomical high efforts to be segmented for compaction or that cannot be treated further because of radiation protection reasons are packed into waste containers and will be imbedded into a fixation matrix by pouring. Typical wastes are large pipes with thin walls and high inner surface contamination, large and difficult to handle parts of tank containers, very long cables, loose debris of building materials or debris of building materials are filled into drums. Facilities for pouring of solid materials are quite simple and may be operated at the site of decommissioning. It has to be assured that the fixation material hardens completely and that in the case of loose debris filled into the waste container (e.g. debris from buildings) the fixation material shows good flow properties. Fixation materials of low viscosity (good flow properties) are produced on the basis of special cementatious mixtures.

The main components of pouring facilities for wastes already being filled into a container are:

- Caisson for emptying debris of building materials or contaminated earth from 200-ltr-drums into a final waste container. The caisson is equipped with special air conditioning devices.
- Mixing, conveying and dosing unit including a tank for dry cement, mixer, pump and tank for liquids that are used for the production of poured mass and serving as transport medium to allow for pumping the solid masses.

6.5.4.3 Direct packaging

Direct packaging of highly activated or contaminated metal parts can be the most advantageous way of optimising radiation protection for the conditioning personnel and storage requirements. By direct packaging segmenting work for larger parts is avoided and thus, in addition to minimising working times, the danger of spreading contamination is substantially lowered. In the case of highly contaminated parts with surface layers which can become easily airborne, packaging before transportation is safer and less time consuming than decontamination work. For a storage cask filled with bagged parts of bulk items, the package density is much lower than that which can be achieved if the parts had been size reduced prior to packaging.

Direct packaging can also be applied for storing debris from concrete or building materials. This method is restricted to low active materials because of the necessity of

manual loading of containers. In addition, harmful and radioactive dust in the air must be carefully controlled and protective measures taken, e.g. breathing masks for the workers. The packing density is not normally satisfactory, but can be easily increased by cementation. One reason for direct packaging is that loose concrete debris or other building materials can not be compacted satisfactorily.
Loose building material debris can also be packed into drums which in turn will be packaged into containers. Gaps and cavities are filled with cement.

6.5.5 Conditioning methods for liquid materials

6.5.5.1 Bituminization

Bituminization of radioactive wastes has been widely used as a conditioning technique and can be applied to a wide range of wastes. However, it is no longer used because of specific developments of final repository requirements. Types of wastes suitable for incorporation into bitumen are chemical sludge, ion exchange material, regenerants and concentrate salt solutions, organic solvents, incinerator ashes, plastic waste and other solid wastes. As in cementation, wastes are embedded into a matrix, so the volume of the final product is in principle increased, however because water can leave the matrix, there may also be some degree of volume reduction.

The basic principles can be summarised as follows: the liquid or solid radioactive wastes are mixed with molten bitumen at a temperature of 110 to 230°C. Water and highly volatile constituents in the heated layer of the bitumen are evaporated. The remaining water-free product is packed into a suitable container ready for disposal. The conditioning of radioactive wastes by bituminization is applied mainly to "liquid" wastes such as filter sludge, ion exchange resins and evaporation concentrates. Bituminization is often applied to wastes that will be disposed of at surface near facilities, so the aspect of leach ability of the product is extremely important.

Optimisation of the properties of the final product is achieved by adjusting the quantities of reagents used according to the behaviour and the quantity of those radio nuclides contained in the liquids. By chemical co-precipitation, decantation and homogenisation, liquid wastes can be decontaminated in such a way as to leave the majority of the activity within the sludge. The volume of sludge can be reduced by decantation. Sludge are homogenised in an extruder supply tank which is the main part of the plant. The process parameters are adjusted according to the results of analyses of samples taken during homogenisation. Using this method a homogeneous mixture of bitumen and wastes is obtained. Effective filling of drums with the final product can be achieved by placing the drums into a carousel. After filling, the drums are cooled for about a day. Normally they are not hermetically sealed in order to allow release of gaseous products. Bitumen is normally resistant to many reagents at atmospheric temperatures.
In the case of storage in deep geological formations, bitumen remains resistant at higher temperatures. However, in principle, chemical processes such as oxidation and microbiological degradation can not be eliminated. The selection of a suitable bitumen is carried out by using a standardised method for the evaluation of the necessary physical properties such as penetration, viscosity etc. Its main disadvantages are its flammability and its low resistance to radiation.

6.5.5.2 Drying Facilities

Liquid wastes and concentrates are converted into solid state by means of mobile or stationary drying facilities. Drying is defined as lowering the water content of the waste far beyond limits that are reached by de-watering (mechanical) techniques. As a type of pre-treatment, it may follow de-watering processes such as evaporation. To avoid the spread of contamination and its related risks, drying is often carried out at very low pressures (vacuum) and by means of heat. It may be regarded as a type of pre-treatment as well as a type of final treatment.
Radioactive wastes may be dried as loose or bulk material that has already been packed into a waste container. Here heating of the total waste package is required. Evaporation of water vapour is facilitated by operation at very low pressure. An examination of a device used to dry loose or bulk materials is a 'drum- dryer'.

Drum Dryer

Drum dryers are used for the continuous drying of evaporation concentrates to powder-like interim products. The concentrate is spread in a very thin layer onto the surface of a drum heated by vapour (see figure 6.7). It sticks to the rotating drum's surface and is finally removed by a scraper. It then falls down into a waste drum and may be subjected to further conditioning techniques such as immobilisation.

Drying with the help of very low pressure and temperature

An example of a mobile plant for drying liquid wastes is the so called FAVORIT plant (see figure 6.8). This is suitable for drying liquid wastes with a solid content of approximately 20% by weight (e.g. evaporator concentrates) filled in waste containers (200-ltr drums). Up to six waste casks can be placed into the heating chambers and can be attached to the central vacuum unit. The evaporation capacity is in the range of 50 to 150 l/hr. The liquid wastes are pumped via the dosing tank into the final storage casks. The resulting product is a monolithic salt block. Heating is performed electrically from the outside while the interior of the casks is held under vacuum. The water vapour is directed through a condenser. The condensate flows through a collecting tank into the waste water sump of the nuclear power-plant.
Another example with a slightly different design is the so called ROBE plant (see figure 6.9). Concentrates from evaporation are dried together for example with filter concentrates. Compared to cementation, a volume reduction factor of 12 can be achieved in the conditioning of spent ion exchanges resins. Drying is performed within a mantled evaporator and homogeneity of the material is achieved by stirring. The dried material flows into the final waste drum where it hardens. In contrast to other products, the product from the ROBE plant has no exact melting point, instead showing a softening interval.

Evaporation facilities

Evaporation facilities are normally stationary, hence radioactive liquid wastes have to be transported for treatment. The solid content of liquid wastes is increased due to

evaporation of about 25% of the weight. This concentrate is either solidified with a flux material (e.g. cementation) or converted into a solid waste product by drying.
Chemical compounds contained in liquid wastes are subjected to different kinds of pre-treatment such as filtration, decantation, neutralisation, precipitation or, in the case of strong acids, dilution. The last option of pre-treatment is, in comparison to the other methods, much easier to perform, but requires a longer treatment time or evaporation facilities on a larger scale.
To typical evaporation facilities are shown in figures 6.10 and 6.11. Throughputs are in the range of 1 to 10 m^3/hr. Evaporation does not result in a final product, but needs additional treatment steps, for example solidification, drying etc. Decontamination factors for evaporation are in the range of 10^4 to 10^5. In most cases the evaporated water content can be released via a normal purification plant to the environment.
A special type of an evaporation facility is a so called thin-film evaporator (see figure 6.12). This type of evaporator is used in order to solidify liquid concentrates from other evaporation facilities, active ion exchange resins (LLW and ILW) mixed with sludge and spent oil and organic solvents in very small quantities.
Concentrates from evaporation facilities are normally pumped into storage vessels with volumes of 7 m^3. From there the concentrates are pumped into the head of the vertically arranged evaporator which is equipped with movable strippers at eight different heights. The concentrates are evenly spread onto the top of the inner surface of the evaporator. Transportation from the top to the bottom is carried out by gravity with the help of the rotating strippers, which are forced against the surface due to the rotation. At the bottom of the evaporator a high-viscous dehydrated product is drained into a waste container (drum). The waste container is tightly connected with the system by a basket.

The product quality can be varied from liquid-aqueous to powder-like solids by the following means:
- the feed rate can be adjusted
- the partial pressure of water vapour can be reduced, the water/water vapour equilibrium can be shifted by maintaining an air input (up to 20m^3/h bypass of air)
- low pressure operation can be used by applying 2 kPa to the head of the evaporator. The specific activity in the condensate is 10^{-3} to 10^{-4} times the specific activity of the liquid evaporator concentrate.

6.5.5.3 Cementation

Facilities for the cementation of liquid wastes can be either mobile or stationary and convert liquid wastes into a homogeneous cement stone block. The technique of cementation can be used for the conditioning of liquid wastes arising during decommissioning but, because of the increased volume, cementation is considered to be less advantageous than drying.
However, cement has a number of advantages as an encapsulation matrix as noted below:
- It requires relatively simple process plant operating at low temperatures.
- It has a high density which provides considerable self-shielding.
- It is highly alkaline and has properties which lower the solubility of radio nuclides.
- It is a low cost option (but a high cost option according to storage costs).

Prior to cementation liquid wastes containing chemicals arising from decontamination processes such as electropolishing or acid treatment may be subjected to neutralisation or precipitation or, in the case of strong acids, they may be subjected to dilution processes. The nature of pre-treatment depends on the particular situation and the effectiveness of the pre-treatment process also has to be considered.

Three types of cementation facilities used for the cementation of evaporation concentrates, sludge and ion exchange resins are briefly described. World-wide, several types of systems have been developed, e.g. in-drum-mixing, external mixing, continuous mixing. All these systems may be grouped according to whether the final product is produced within the final waste package or whether it needs to be transferred from the place of production into a waste package. Alternatively they may be grouped according to the mode of operation: batchwise or continuously. Firstly, a mobile plant used widely in Germany is described.

In-drum-mixing

The mobile plant, called FAFNIR, works either with a tumbler or a stirring mechanism (see figure 6.13). The tumbler version is the most commonly used. It can be operated with either wet or dry cementation. Wet cementation is used for raw liquid wastes. The binding agent is placed in the waste cask as dry material. Following this, the liquid wastes are placed in the cask. Alternatively, raw dried wastes are placed into a waste cask and immobilisation is achieved by adding the cementatious material. Filling and mixing are performed in a similar way for both modes. In the case of wet cementation the process can be summarised as follows: Sludge is either sucked or pumped into the dosing tank. The waste cask is evacuated and then rotated and tipped (tumbled). Sludge is then sucked into the cask and mixing is commenced. After sufficient time, tumbling is stopped and the central opening of the lid of the cask is sealed.

Normally, 200-ltr-drums are used as waste casks with a central opening on the top of diameter 50-80 mm. The drum is equipped with mixing paddles. After filling and mixing the drum contents are allowed to cure. Typical throughputs are about 1 m^3/hr.

External mixing

The facility described above can also be operated with external mixing. Here the wastes are mixed outside the final waste package, but are transferred directly after mixing into the final waste container. Figure 6.14 summarises this process.

Continuous mixing

This kind of facility is used for the cementation of solid alpha- bearing wastes using contaminated liquids as the binding water in the cement. Shredded organic wastes are mixed with cement and liquid wastes and form a free-flowing product that can be poured and used to encapsulate solid organic wastes.

6.5.5.4 Incineration

Liquid wastes with a high content of organic material can be incinerated in specially designed plants or in plants adapted from solid wastes incineration. Organic liquid waste materials arise during decontamination from removal of coatings, paint or by using oils. Contaminated oils arise as primary waste in pumps, transformers etc. Heating requirements of the organic wastes are up to 40,000 kJ/kg and throughputs are in the range of 10 to 40 kg/hr.

6.6 References

[6.1.] IAEA, Radiological Characterisation of Shut Down Nuclear Reactors for Decommissioning Purposes, Technical Reports Series No. 389, IAEA, Vienna, 1998.

[6.2.] IAEA, Cleanup and Decommissioning of a Nuclear Reactor after a Severe Accident, Technical Reports Series No. 346, IAEA, Vienna, 1992.

[6.3.] IAEA, Design and Construction of Nuclear Power Plants to Facilitate Decommissioning, Technical Reports Series No. 382, IAEA, Vienna, 1997

[6.4.] Bach et.all.
Handbook on decommissioning of nuclear installations
EUR 16211, 1995

Chapter 6

Annex

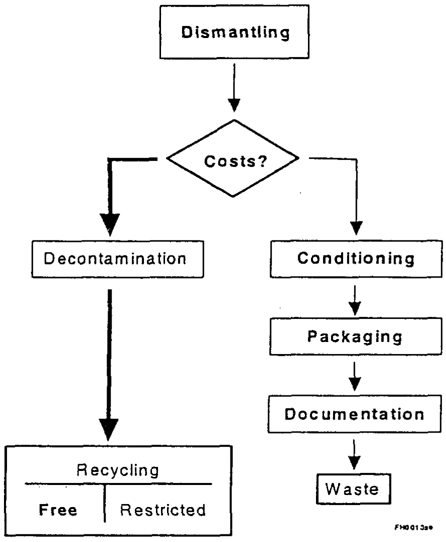


Fig. 6.1: Factors for waste management

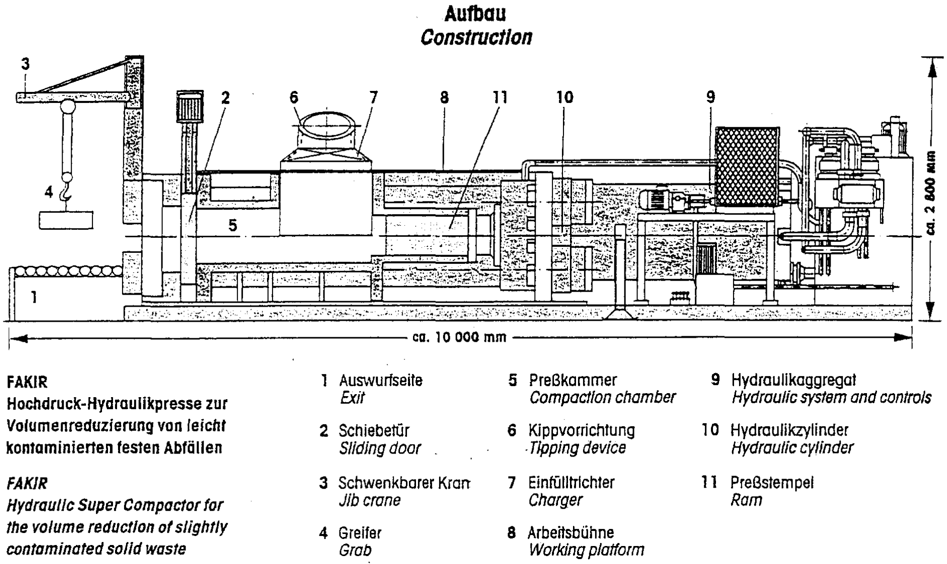


Fig. 6.2: The FAKIR - Supercompactor

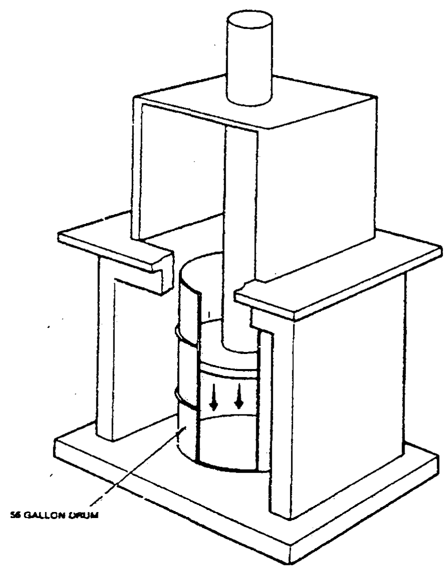


Fig. 6.3: Schematic drawing of an In-drum compactor

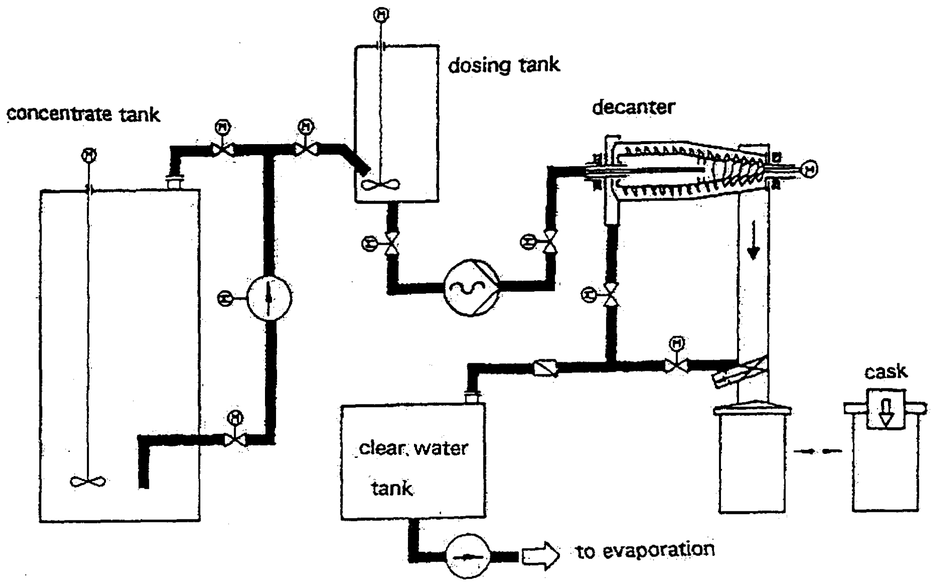


Fig. 6.4: Implementation of an In-drum compactor in a treatment line

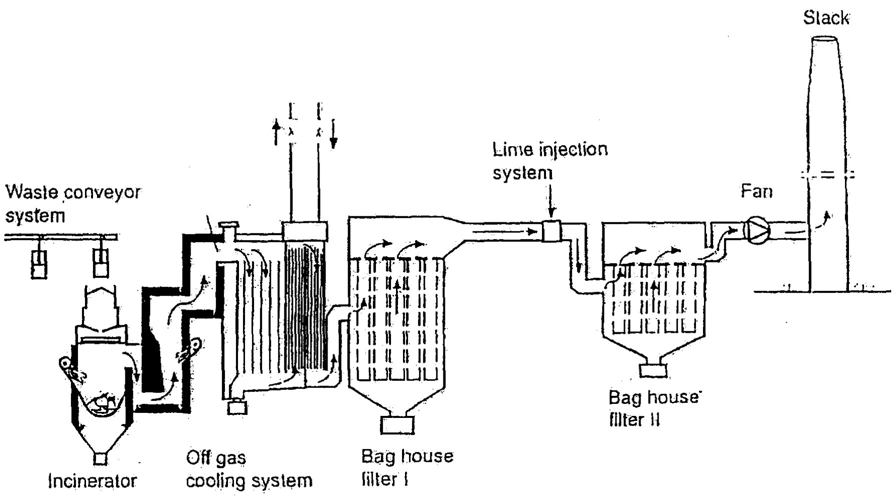


Fig. 6.5: Shaft Kiln with tippable grate

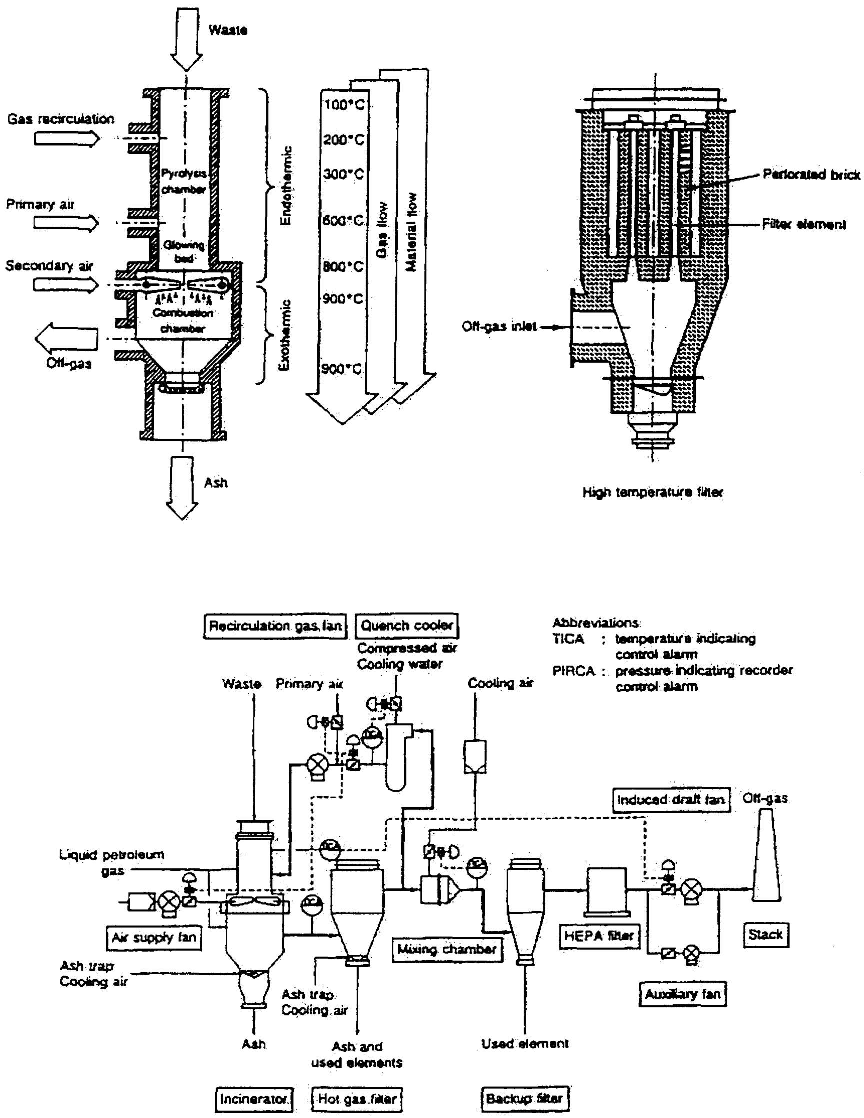


Fig. 6.6: Shaft Kiln with movable grate

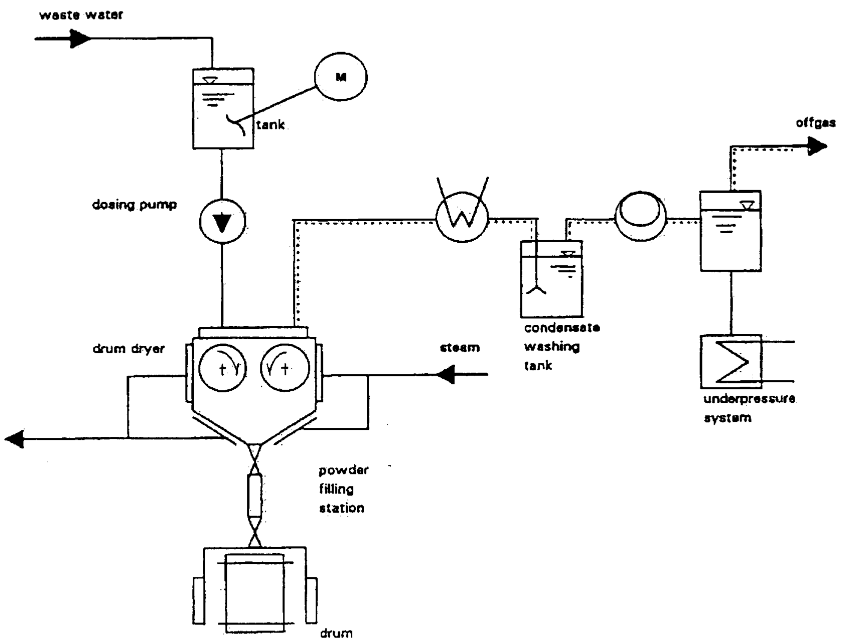


Fig. 6.7: Schematic drawing of a drum dryer system

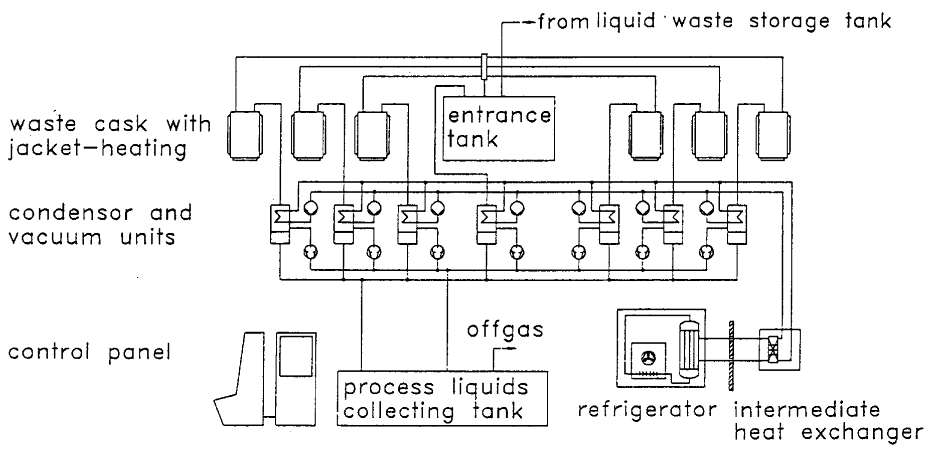


Fig. 6.8: Scheme of the FAVORIT plant

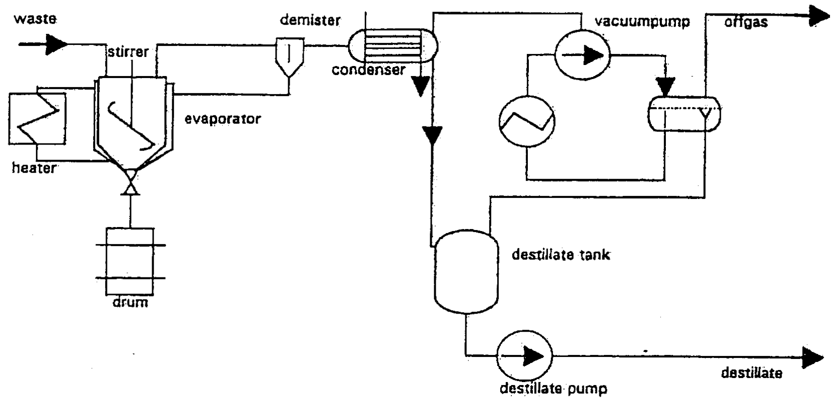


Fig. 6.9: Schematic drawing of the ROBE process

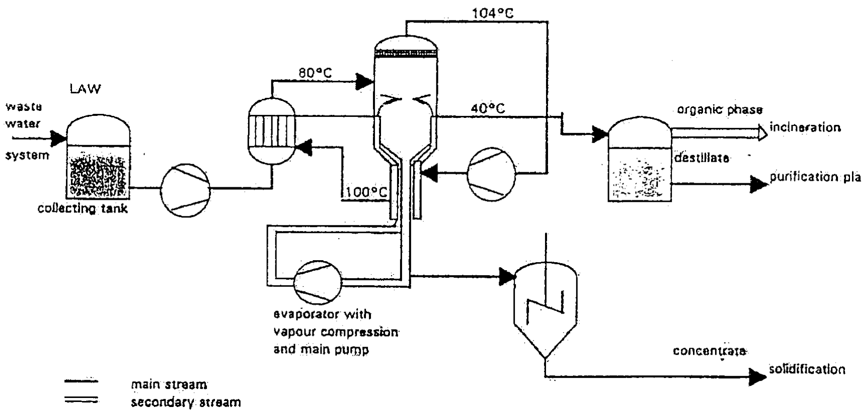


Fig. 6.10: Evaporation plant for low level waste (1)

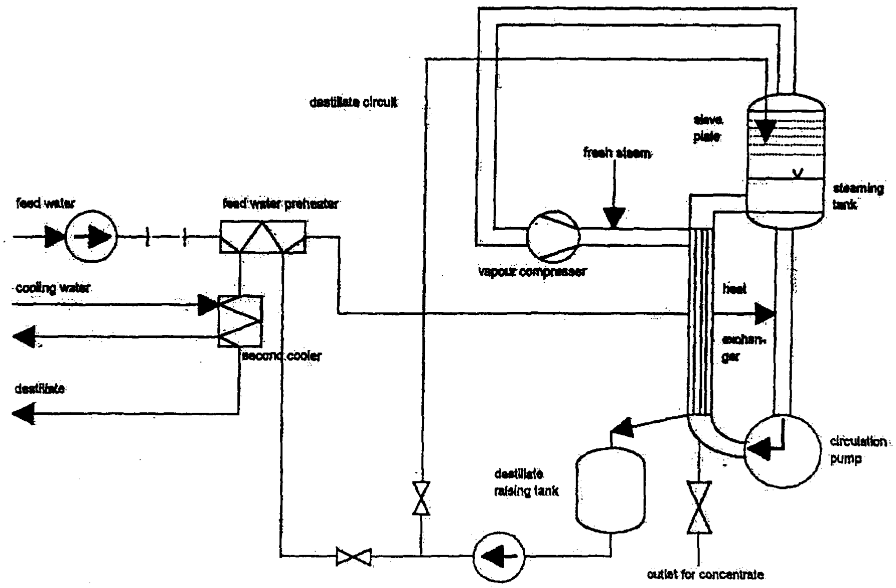


Fig. 6.11: Evaporation plant for low level waste (2)

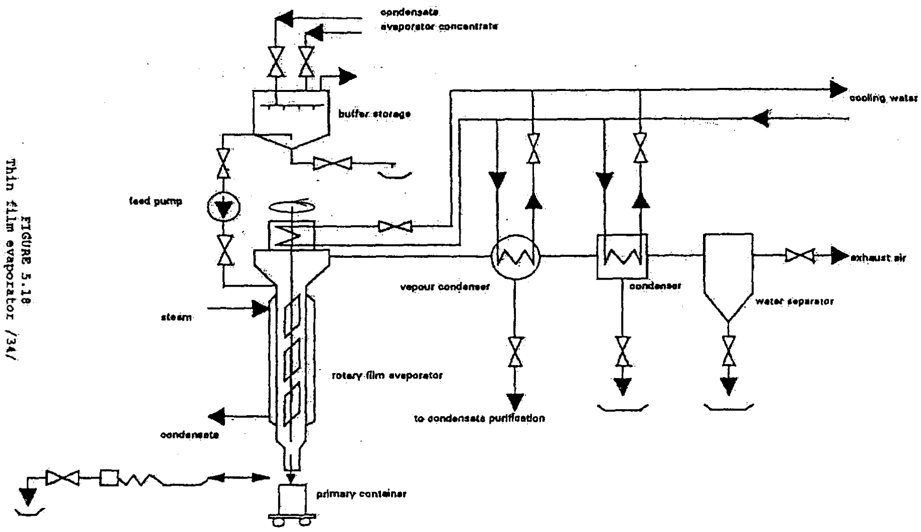
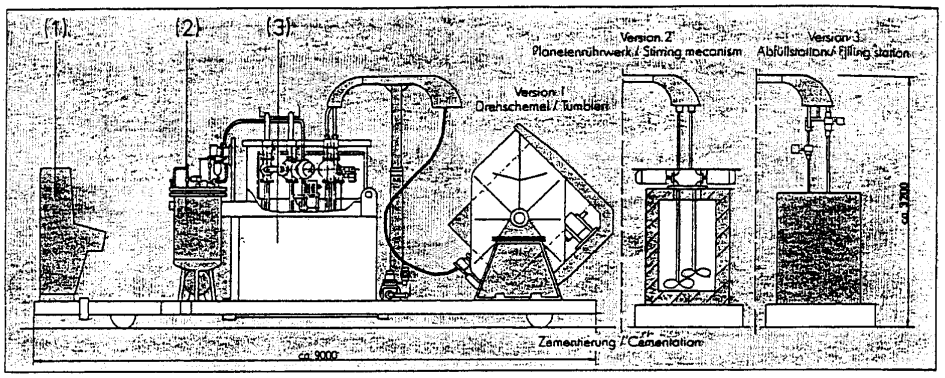


Fig. 6.12: Thin fill evaporator



FAFNIR
Mobile Konditionierungsanlage für radioaktive Flüssigabfälle und Ionentauscherharze

FAFNIR
Mobile conditioning facility for radioactive liquid wastes and ion exchange resins

(1) Schalt- und Steuerpult / Switch and control board
(2) Vakuumvorlagebehälter / Vacuum container
(3) Dosierbehälter / Dosing tank

Abbildung Titelseite: FAMA-Anlage
Photograph on front page: FAMA facility

Abkürzungen / Abbreviations: m.t.: Metric tonnes

Hauptkenndaten / Main Features	
Abmessungen/Dimensions:	Länge ca. / Length approx. 9000 mm
	Breite ca. / Width approx. 2500 mm
	Höhe ca. / Height approx. 3200 mm
Gewicht / Weight:	ca. / approx. 20 t (m.t.)
Raumbedarf / Space requirement:	ca. / approx. 11 x 5 m
Strombedarf / Power consumption:	max. 7.5 kW
Kapazität / Capacity:	bis zu 15 Abfallbehälter pro Tag /
	up to 15 waste containers per day

Fig. 6.13: FAFNIR cementation plant

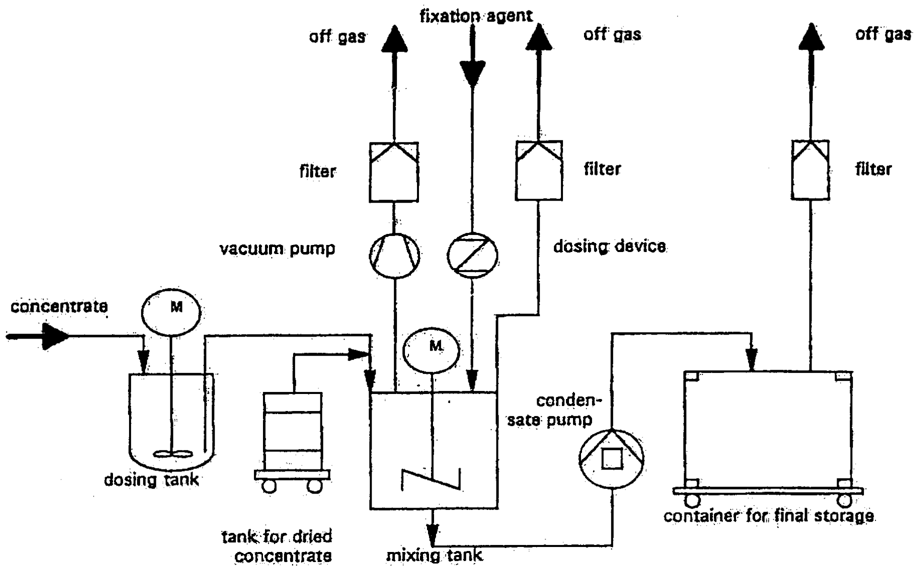


Fig. 6.14: FAFNIR plant for external mlixing, simplified flow sheet

Appendix 1

THE MAIN PROBLEMS OF DECOMMISSIONING OF NUCLEAR AND RADIATION HAZARDOUS INSTALLATIONS IN RUSSIAN RESEARCH CENTRE "KURCHATOV INSTITUTE"

Authors: E.P. Ryazantsev, V.I. Kolyadin, B.K. Bylkin, Yu.A. Zverkov
(RRC "Kurchatov Institute")

At the present time the Russian Research Centre "Kurchatov Institute" (RRC "KI") being created 57 years ago for the development and realization of atomic project, is the biggest research centre of Russia. There are the specialized physical experimental installations of different type, including the research experimental reactors, constituting the main base of experimental facility for nuclear and neutron physics, physics of solid materials and superconductivity, for atomic energetic and other directions of atomic science and techniques.
It is here, in RRC "KI", in the former Soviet Union, in December 1946 the first research reactor has started up its operations and at the first time on the continent of Europe and Asia, the chain reaction of decay of uranium has been realized.

While at the beginning the facility has been provided for RRC "KI" in the surroundings of Moscow, now it occurred on the territory of the town due to decades of intense urban development.

This historically developed situation is the characteristic example of situations with a big nuclear centre located on the territory of big town, when the sanitary-protective zone of complex reactor experimental base, limited by perimeter of territory of this centre, locate at the neighboring of 100 meters from the habitude town building. (Fig. 1).

The overall square of territory of RRC "KI" is around 100 ha, including 4 ha of square of independent territory with experimental complex in the frame of 2 research nuclear reactors. At the recent time on the territory of RRC "KI" there are 9 research nuclear reactors (Table 1), part of which have been already stopped, 17 critical assemblies, "hot" material study laboratory, as well as sites of storage of nuclear material and radioactive waste.

The inevitable result of activities of experimental facility of RRC "KI in the frame of scientific research on realization of military and civil programs in the field of atomic energy during period 1943-2000, is the accumulation of spent nuclear fuel (SNF) and radioactive waste (RAW) on its territory.

The summary activity of SNF, accumulating at the storage facilities of RRC "KI" exceeds $1x10^{17}$ Bk. Around 900 peaces of thermo-irradiating elements have been accumulated there by overall mass around 6 ton.

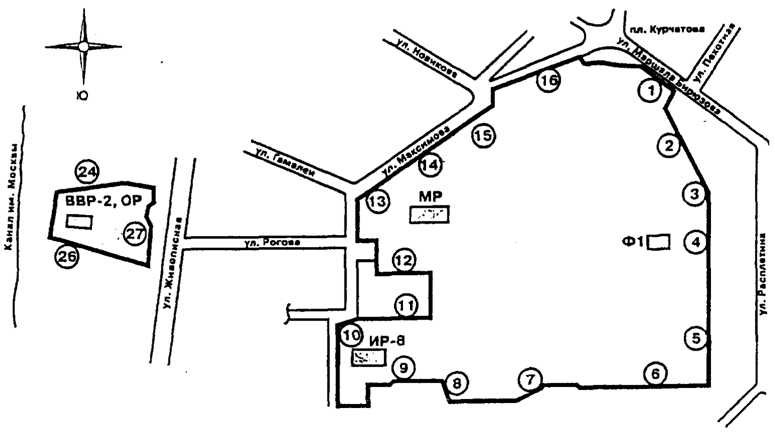


Fig. 1. The scheme of territory of RRC "Kurchatov Institute" and sites of control of doses rate of gamma-irradiation on its perimeter

The summary activity of accumulated RAW, located at the interim storage facilities of RRC "KI", is around 1200 cubic meters, and their mass is evaluated as 2000 ton, except the contaminated grounds.

The RAW is located in 10 interim storage facilities, which occupy the square more then 2 ha.

The storage of wastes in these storage facilities have been started at the beginning of 50-s years. In 1965, after organization of enterprise SPA "Radon", created following the title of RRC "KI", the Low Level Radioactive Waste have been sent to the storage facilities of this enterprise, and in 1972, since the operation of storage facility of High Level Radioactive Waste on this enterprise, the regular removal of these radioactive waste for disposal at SPA "Radon" have been arranged.

The square of ground on the territory of temporary storage facility of RAW is radioactively contaminated. The radionuclides of Co-60, Sr-90, Cs-134, Cs-137, Eu-152, Americium-241 have been found in the samples of ground.

Nevertheless, at the present time, the radiological situation on the territory of RRC "KI" in general could be considered as satisfied enough.

It is justified by the fact data of long-term control of radioactivity arriving to the environment through the systems of ventilation of reactor installations existing in RRC "KI", and of the state of elements of this environment during last decade.

Table1. Research nuclear reactors of RRC "Kurchatov Institute"

N	Title	Type of reactor	Thermal power Project MWt	Termal power After recons-truction MWt	Time of start-up	State
1	F-1	Uranium-graphite	0.024		1946	operating
2	WWR-2	Water-Moderated Water-cooled Tank	0.3	3	1954	stopped and dismantled in 1983
3	RTF	Channel, Graphite	10	20	1952	dismantled in 1962
4	MR	Channel, loading in the basin	20	40	1963	stopped in 1993
5	IRT	Basin	2	8	1957	stopped and dismantled in 1979
6	IR-8	Basin	8		1981	operating
7	GIDRA (IIN-3M)	Homogeneous solution, impulse	0.01-30 MJ in impulse		1972	operating
8	GAMMA	Vessel Water-moderated water-cooled	0.125		1982	operating
9	ARGUS	Homogeneous Solution	0.02		1981	operating
10	OR	Water-moderated water-cooled Tank	0.3		1989	operating
11	ROMASHKA	High temperature breed neutrons with thermo-electrical trasformator	0.04		1964	Stopped and dismantled in 1966

One may notice, that the fallout of radionuclides in atmosphere during all time of existence of RRC "KI" have been in the frame of permitted sanitary normative and as rule, have been sufficiently low then the permitted limits.

During already a few years, there is the systematic monitoring in RRC "KI" of the external gamma-phone (dose gamma-irradiation) of some control sites, located over the perimeter of its territory (Fig.1).

The data of this monitoring are regularly published. The results of this monitoring confirm that the average value of external phone does not differ from analogous value in other regions of Moscow.

Nevertheless the location of such big nuclear centre on the territory of such megalopolis as Moscow, certainly create situation, which demand scrupulous analysis and control.

After the Chernobyl accident and strengthening of requirements of normative documentation this neighboring objectively provoke the increased anxiety and attention of town authorities, society, and population.

At the present time this attention is expressing also towards acceleration of works on preparation to the decommissioning of already stopped research reactors MR, WWR-2 and "ROMASHKA", as well as of works dealing with liquidation of existing sites of storage of nuclear and radioactive materials and rehabilitation of sites of its disposal.

At the same time the development of works in RRC "KI" on preparation of decommissioning of stopped research reactors and on removal of storage facilities of SNM and RAW from its territory is complicated by a few factors.

Namely, the planning of works on decommissioning of research reactors requires an accounting of multiformity of the construction differences, complicated the development of unified technical solutions; high density of flows of neutrons in some reactors, exceeding the density of flow of neutrons of energetic reactors; what determine the high levels of induced activity in the constructions; that in the content of some reactors there is a big amount of experimental installations (loop constructions, being the analogues of main elements of constructions of Nuclear Energetic Installations of different purpose), what requires detailed development of dismantling technologies for carrying our the works; that these reactors; as rules, functioning not independently, but exploiting in the frame of scientific-experimental complexes, including also other radiation-hazardous subjects (material science "hot" laboratories, critical assemblies, accelerators of charged particles etc.) that these reactors are located in the nearest place from high-density living territory, what underline the increased requirements towards provision of safety of carrying out the works;

Two following technical problems complicate the solution of tasks on preparation of decommissioning and decommissioning of research reactors of different types: the necessity to develop efficient and economical technologies for carrying out the deactivation of equipment, pipe-lines and buildings in connection with absence on the scientific reactors of regular systems of deactivation; the necessity to develop the specialized means of technological provision, required for the carrying out the dismantling and fragmentation of big amount and different nomenclature of radioactive equipment, systems and pipe-lines.

In fact, the choice of strategy of decommissioning of research reactors and carrying out the concrete works on them might depend on the solving of problem with development of above-mentioned technologies.

Beside of this, at the present time the development of works on planning of decommissioning of research reactors in RRC "KI" is preventing by the follow additive factors, which are the general for Russia: during period of exploitation on the research reactors a huge amount of experimental thermo-emitting assemblies have been accumulated, which requires a special technologies for treatment due to the "non-standard" of content of fuel compositions; centralized sites for long-term storage of RAW of different categories of activity and spent nuclear fuel are absent.

The normative base on regulation of decommissioning of nuclear installations, as research reactors, is only at the initial stage of its development.

The process of planning of decommissioning and decommissioning of research reactors in Russia still has not the mass character and till the present time no one research reactor, which have been completely stopped, could be considered as that decommissioned till the Stage 3 (state of "green" field, in accordance with classification of IAEA).Unfortunately, it is necessary to notice, that till the present time the national policy in Russia on decommissioning of research reactors still have not been developed at full extent and financially have not been supported due to different difficulties.

Indeed, one may to consider, that in Russia and in RRC 'KI" in particular, the first experience have been acquired on carrying out the works on dismantling of research reactors. Nevertheless this experience deal with the realization of reconstruction of some "ancient" reactor installations, determined by the requirements to the enlargement of their neutron-physical parameters, as well as by work out of recourse of its equipment.

In some cases the works on reconstruction of research reactors include the full montage of all internal corpus (internal reactor) installations. For example, at RRC "KI" in 1962, after decades of exploitation the research reactor RFT have been stopped. After removal of nuclear fuel the dismantling of equipment have been realized from the mine of reactor, after what the mine have been solidified in concrete, while the reactor facility and a part of its equipment have been used for creation of new reactor MR.

As concerns evacuation of ammunition and liquidation of sites for temporary storage of RAW located on the territory of RRC "KI", the fulfilling of these works is one of the central problems for RRC "KI" for the nearest time, including that in the field of improvement of radiological situation and safety of nature on its territory.

Nevertheless the solution of problems of removal of RAW and rehabilitation of sites of it temporary storage for RRC "KI" is complicated by big volumes of accumulated wastes and impossibility to remove them for disposal to the enterprises of SIA "Radon" due to limits of existing possibilities of this enterprise.

For the full rehabilitation of sites for storage of RAW on the territory of RRC "KI" around 4 thousands cubic meters of contaminated grounds is required to remove additively.

So far, the preparation of decommissioning and decommissioning of nuclear- and radiation-hazardous installations of RRC "KI" are sufficiently difficult and multi-aspect problem, which requires the complex approach and solving of which requires the long time, significant labor, material and financial resources.

EUNDETRAF

Chapter 7

Waste Characterization and Measuring Methods

Chapter summary

Waste Characterization and Measuring Methods

7.1 *Waste characterization*

7.1.1 Waste classification

As shown in chapter 6, the decommissioning of nuclear facilities generates a large volume of materials of very different kinds (concrete rubble, metals, thermal insulation, electrical cables ...). Therefore, in order to ease the definition of optimal waste management schemes (from waste treatment/conditioning to final disposal), it is necessary to classify all these types of waste.

As part of its RADWASS programme the IAEA has issued a safety guidance document, Classification of Radioactive Waste [7.1], which revises and updates previous publications on the subject. In section 2 of the document, the purpose of and the approach to radioactive waste classification is presented. The IAEA defines the term 'classification' as "an approach which is used, mainly when the quantity of elements considered (objects or ideas) is large, to ease management of the elements by reducing their number". Classification is realised by selecting the main features (criteria) and by structuring these criteria.

Classification may be more or less precise depending on the number of classes and the criteria considered. The degree of differentiation depends on the purpose of the classification. For radioactive waste, classification systems may be derived from different points of view, such as safety related aspects, process engineering demands or regulatory issues. Adopting a classification system will serve many purposes at all stages (e.g. conditioning, interim storage, transportation) from waste generation to eventual disposal.

An ideal radioactive waste classification system should meet, but not be limited to, the objectives listed below:

1. cover the full range of radioactive waste types;
2. address all stages of radioactive waste management;
3. relate radioactive waste classes to the associated potential hazard;
4. be flexible to serve specific needs;
5. change already accepted terminology as little as possible;
6. be simple and easy to understand; and
7. be as universally applicable as possible.

Waste classification systems can be based on purely qualitative or quantitative criteria or a combination of both. Examples of qualitative criteria would be waste origin (e.g. reactor, reprocessing plant) and physical form (e.g. solid, liquid, gas). Surface dose rates and activity concentrations are examples of quantitative criteria.

There are numerous parameters or properties that can be used to classify radioactive waste, some of which are listed below.

1. Origin
2. Criticality
3. Radiological properties:
   - half-life
   - heat generation
   - intensity of penetrating radiation
   - activity and concentration of radionuclides
   - surface contamination
   - dose factors of relevant radionuclides

4. Other physical properties:
   - physical state (solid, liquid or gaseous)
   - size and weight
   - compactability
   - dispersibility
   - volatility
   - solubility
   - miscibility
5. Chemical properties:
   - potential chemical hazard
   - corrosion resistance/corrosiveness
   - organic content
   - combustibility
   - reactivity
   - gas generation
   - sorption of radionuclides
6. Biological properties:
   - potential biological hazards

The act of describing any of these waste properties using analytical tools and/or techniques is defined as waste characterisation. Various invasive, non-invasive, destructive and non-destructive techniques have been developed and used by the nuclear industry to characterise radioactive waste.

The IAEA in previous publications proposed a radioactive waste classification system based on 3 classes (categories) of waste namely:

1. High Level Waste (HLW);
2. Intermediate Level Waste (ILW); and
3. Low Level Waste (LLW).

The definitions of the 3 waste categories are given below.

<u>High level waste</u>

(i) The highly radioactive liquid, containing mainly fission products, as well as some actinides, which is separated during chemical reprocessing of irradiated fuel (aqueous waste from the first solvent extraction cycle and those waste streams combined with it).
(ii) Any other waste with radioactivity levels intense enough to generate significant quantities of heat by the radioactive decay process.
(iii) Spent reactor fuel, if it is declared a waste.

<u>Intermediate level waste (medium level waste)</u>

Waste which, because of its radionuclide content requires shielding but needs little or no provision for heat dissipation during its handling and transportation.

<u>Low level waste</u>

Waste which, because of its low radionuclide content, does not require shielding during normal handling and transportation.

Within the ILW and LLW classification, the IAEA also differentiated between short and long lived waste, as well as alpha bearing waste. Here the term short lived waste refers to radioactive waste which will decay to an activity level which is considered to be acceptably low from a radiological viewpoint, within a time period during which administrative controls can be expected to last. Long lived waste is radioactive waste that will not decay to an acceptable activity level during the time which administrative controls can be expected to last.

Alpha bearing waste is radioactive waste containing one or more alpha emitting radionuclides, usually actinides, in quantities above acceptable limits established by the national regulatory body.

This earlier classification system had no clear linkage to safety aspects in radioactive waste management, especially disposal. Additionally, the system lacked quantitative boundaries between classes and recognition of a class of waste that contained so little radioactive material it could be exempted from control as radioactive waste. Finally, it lacked recognition of wastes such as those from mining and milling uranium ore, which contain small quantities of natural radionuclides dispersed through large volumes of material.

To address these limitations the IAEA has proposed a modified classification system [1] based on three major waste classes. The waste classes are described below.

1. Waste containing such a low concentration of radionuclides that it can be exempted from nuclear regulatory control in accordance with clearance levels, as the associated radiological hazards are negligible.
2. Waste that contains such an amount of radioactive material that it requires actions to ensure the protection of workers and the public either for short or for long periods of time. This class covers a very wide range of radioactive wastes, ranging from radioactive waste just above exempt levels, e.g. not requiring shielding or particular confinement, to radioactive waste that contains such high levels of radioactivity that shielding and possibly cooling may be required. A range of disposal methods may be postulated for such waste.
3. Waste that contains such high levels of radioactive material that a high degree of isolation, normally geological isolation, from the biosphere is required over long time periods. Such waste normally requires both shielding and cooling.

The principles of the earlier classification system have been retained, however, the revised system has been organised to take account of safety disposal issues. The revised waste classification system defines 3 principal waste categories namely:

1. Exempted Waste (EW);
2. Low and Intermediate Level Waste (LILW); and
3. High Level Waste (HLW).

Experience within the nuclear industry world-wide has allowed the IAEA to suggest quantitative boundaries between the categories for solid radioactive waste. More detailed quantitative boundaries could be developed in accordance with national programmes and requirements. The quantitative boundaries for each waste category are reported in Table 1 along with possible disposal options. The definition, of the short lived waste (LILW-SL) provides guidance on the expected limits on alpha emitting radionuclides to 4000Bq/g in individual packages and an overall average of 400Bq/g per waste package for disposal to a near surface repository.

Table 1
IAEA Radioactive waste classification

Waste classes	Typical characteristics	Disposal options
Exempted Waste (EW)	Activity levels at or below clearance levels which are based on an annual dose to members of the public of less than 0.01 mSv	No radiological restrictions
Low and Intermediate Level Waste (LILW)	Activity levels above clearance levels and thermal power below about 2 kW/m^3	
Short Lived Waste (LILW-SL)	Restricted long lived radionuclide concentrations (limitation of long lived alpha emitting radionuclides to 4000 Bq/g in individual waste packages and to an overall average of 400 Bq/g per waste package)	Near surface or geological disposal facility
Long Lived Waste (LILW-LL)	Long lived radionuclide concentrations exceeding limitations for short lived waste	Geological disposal facility
High Level Waste (HLW)	Thermal power above about 2 kW/m^3 and long lived radionuclide concentrations exceeding limitations for short lived waste	Geological disposal facility

The principles related to waste exemption and clearance levels are detailed in paragraph 7.2 of this chapter, which also gives some values of clearance levels for radionuclides such as Co-60 and Cs-137.

7.1.2 Waste acceptance criteria

7.1.2.1 Basic Safety Principles

As a first step in the development of Waste Acceptance Criteria (WAC), it is important to establish the internationally accepted fundamental safety principles which underpin all activities and facility operations associated with radioactive waste management.
Since its foundation in 1957, The International Atomic Energy Agency (IAEA) has provided guidance on the safe management of radioactive waste management. More recently the IAEA has initiated its Radioactive Waste Safety Standards (RADWASS)

programme which is aimed at establishing a coherent and comprehensive set of principles and standards for the management of waste and formulating the guidelines necessary for their application. The RADWASS publications will provide Member States with a comprehensive series of internationally agreed documents to assist in the derivation of, and to compliment national criteria, standards and practices. Publications issued under this programme will revise and update some earlier references on radioactive waste management.
The RADWASS scheme consists of a four level hierarchy of publications with a Safety Fundamentals document at the highest level followed by Safety Standards, Safety Guides and Safety Practices at the subsequent levels.
In the safety fundamentals document "The Principles of Radioactive Waste Management" [7.2] the IAEA defines the objective of radioactive waste management and sets out 9 internationally agreed principles. The overall objective is defined as follows:
"The objective of radioactive waste management is to deal with radioactive waste in a manner that protects human health and the environment now and in the future without imposing undue burdens on future generations."

The 9 principles are listed below:

Principle 1: Protection of human health
Radioactive waste shall be managed in such a way as to secure an acceptable level of protection for human health.

Principle 2: Protection of the environment
Radioactive waste shall be managed in such a way as to provide an acceptable level of protection of the environment.

Principle 3: Protection beyond national borders
Radioactive waste shall be managed in such a way as to assure that possible effects on human health and the environment beyond national borders will be taken into account.

Principle 4: Protection of future generations
Radioactive waste shall be managed in such a way that predicted impacts on the health of future generations will not be greater than relevant levels of impact that are acceptable today.

Principle 5: Burdens on future generations
Radioactive waste shall be managed in such a way that will not impose undue burdens on future generations.

Principle 6: National legal framework
Radioactive waste shall be managed within an appropriate national legal framework including clear allocation of responsibilities and provision for independent regulatory functions.

Principle 7: Control of radioactive waste generation
Generation of radioactive waste shall be kept to the minimum practicable.

Principle 8: Radioactive waste generation and management interdependencies
Interdependencies among all steps in radioactive waste generation and management shall be appropriately taken into account.

Principle 9: Safety of facilities
The safety of facilities for radioactive waste management shall be appropriately assured during their lifetime.

The required performance of a radioactive waste repository is therefore that radioactive waste should be disposed of in a manner that ensures that there are no unacceptable radiological consequences at present and in the future.
The management of radioactive waste must be performed so as to secure an acceptable level of protection for human health. The basic principles for radiological protection are those prescribed by the International Commission on Radiological Protection [7.3] and expanded by the IAEA Safety Series document No.115 [7.4].
The safety of the radioactive waste should be ensured for its lifetime. Therefore a comprehensive safety assessment must be performed and take account of:

- interim storage for decay of radionuclides
- selection of techniques for conditioning radioactive waste
- engineering for handling waste packages
- engineered barriers
- natural barriers
- institutional control period
- administrative methods

The various pathways by which radionuclides might be released from the repository and reach the human environment must be assessed and the radiological consequences quantified. The performance of the repository must then be discussed with the regulatory authorities, and the operator, to determine whether the performance is acceptable and to optimise the performance as required.
Safety analyses are performed to demonstrate that the potential risks do not exceed the limits prescribed by the regulations to protect the human environment during the operational period, the institutional control period and the unrestricted site access period.
A number of post-closure performance criteria can be used. These include the following:

- Risk target
- ALARP principle
- Natural disruptive events
- Human intrusion
- Site selection
- Comparison with levels of naturally occurring radionuclides
- Safety assessment
- Multi-barrier approach
- Timeframes

7.1.2.2 Development of Generic WAC Methodology

A generic WAC methodology which has been developed based on IAEA guidelines and taking account of the best international practices is represented diagrammatically in the Figure 7.1. below.

The key points to recognise are that:

- The implementation of the methodology requires that a national framework for radioactive waste management is in place and the responsibilities of the different parties involved are clearly identified.
- The proposed methodology recognises development of the WAC in three stages:
  - Preliminary WAC based on a non-site specific conceptual repository design
  - WAC development on the basis of a site specific detailed repository design
  - Finalisation of WAC and subsequent evolution based on operational experience and possible changes in regulatory requirements
- The interdependencies among all the steps in radioactive waste generation and management from production to final disposal and beyond must be taken into account. The development of the WAC is dependent on inputs from all aspects of waste management and is an iterative process which parallels development of the waste management route from the generic approach to concept specific approach and finally the site/facility specific approach.
- The methodology recognises that the steps in waste management can occur on different timescales. In the absence of the final disposal route, the waste should be processed according to Waste Package Specifications that should anticipate eventual WAC, and at least meet the safety assessments and other constraints for transport and/or storage.
- In the absence of a repository or a site specific detailed repository design, Waste Package Specifications and/or Preliminary WAC should be 'conservative' to enable early production of waste packages by the waste producers. This approach will ensure that wherever possible these waste packages will ultimately meet the finalised WAC.

7.1.2.3 Identification of Generic Waste Acceptance Criteria

The key inputs for development of generic WAC are as follows:

- International Regulations
- IAEA guidance
- LILW disposal best practice
- Generic waste management scenario

On the basis of IAEA guidance [7.5-7.7], international best practice and operational experience, a series of key waste package characteristics or parameters relevant to waste disposal system performance can be identified which lead to potential need for criteria development. These characteristics are listed in Table 2, and their relative importance with respect to the operation and post closure phases of the repository life cycle are identified.

In addition to the waste package characteristics given in Table 2, other areas that should be considered include:

- Waste package identification
- Waste package configuration (dimensions and weight)
- Responsibilities and organisation
- Quality control and compliance including:
  - control of conditioning process
  - checks on waste packages
  - disposal record and data recording
  - compliance with codes and standards

In the following section (7.1.2.4.), sets of generic waste acceptance criteria derived from the waste package characteristics given in Table 2 are presented. Guidance on the development of each of these generic criteria to facility specific WAC is provided.

Table 2
Applicability of waste package characteristics

Characteristic	Operational period	Post-closure period
Radionuclide content	+	+
Dose rate	+	-
Surface contamination	+	-
Structural stability	+	(+) or +
Chemical durability	(+)	+
Corrosion resistance	+	+
Thermal and radiation stability	(+)	(+)
Combustibility	+	(+)
Gas generation	+	(+) or +
Microbial degradation	(+)	(+) or +
Presence of free liquids	(+)	(+) or +
Presence of chelating or complexing agents	(+)	+
Explosivity and presence of compressed gases	+	+
Pyrophoricity	+	+
Toxicity and corrosivity	+	+
Criticality safety	(+)	(+)

\+ Applicable
(+) Limited applicability
\- Unlikely to be applicable

7.1.2.4 Development of Facility Specific Waste Acceptance Criteria

The important inputs in the development of facility specific WAC are as follows:

- National and International Regulations
- IAEA guidance
- LILW disposal best practice
- Generic WAC
- Actual waste management scenario
- Facility management requirements
- Facility safety assessment.

More specifically within the context of the actual waste management scenario, it is essential that reliable waste characterisation data are available on the wastes to be consigned to the repository. In most cases this will entail sampling and characterisation of the actual wastes themselves.

Before progressing to the development of individual criteria, it is useful to identify a number of general questions to consider:

- Are quantitative or specific requirements imposed by regulation or IAEA best practice, for example, total alpha limit on radionuclide inventory?
- Are quantitative package limits or boundaries already defined in the safety assessment assumptions?
- Is the criterion required to fulfil a basic or fundamental requirement or will some flexibility be allowed e.g. through facility management systems, consideration of exceptions on a case by case basis?
- Is detailed guidance available to develop criteria e.g. IAEA expert panel documentation?
- Can compliance with the criteria be demonstrated in practice?

The following sections 7.1.2.4.1. to 7.1.2.4.19 comprise a discussion of the methods, arguments or ideas that might be used to develop specific WAC for a repository based on the surface or shallow-burial concept. The discussion firstly deals in turn with each of the 16 key characteristics of a waste package that have been derived previously. The characteristics considered are as follows:

- radionuclide content
- dose rate
- surface contamination
- structural stability
- chemical durability
- corrosion resistance
- thermal and radiation stability
- combustibility
- gas generation
- microbial degradation
- presence of free liquids
- presence of chelating or complexing agents
- explosivity and presence of compressed gases
- pyrophoricity
- toxicity and corrosivity
- criticality safety.

In Section 7.1.2.4.20 criteria not related to waste package performance are considered. These include:

- physical dimensions and weight
- unique identification
- responsibilities and organisation
- control of conditioning processes
- checking of waste packages
- disposal record and data recording
- disposal facilities management.

7.1.2.4.1 General Methods for Derivation

The demonstration that a repository for radioactive waste does not give rise to unacceptable risk will be based upon a number of safety cases and performance assessments. The quantification of potential risk in such cases and assessments should be based upon appropriate assumptions regarding the characteristics of the components that comprise the repository and its environment and the evolution of these components over time.

It is expected that most or all WAC ultimately will be derived from the assumptions made in the safety cases and performance assessments regarding characteristics of the waste packages and their evolution over time. In particular, the WAC should be designed such that the actual characteristics of the packages are consistent with those assumed. This will ensure that the assessed risk is a valid assessment of the actual performance of the repository.

The description of the derivation of the WAC assumes that both an 'operational safety case' and a 'post-closure performance assessment' will be available. This documentation is assumed respectively to describe the operation of the repository and the expected post-closure performance. It is recognised that in practice different terminology may be used and that each set of documentation will comprise more than one actual document. Extensive guidance on the approach and methodologies for safety case assessments is available in IAEA literature [7.8-7.13]. Further rationalisation and development of these safety assessment methodologies is currently being addressed within the IAEA's ISAM programme [7.10].

7.1.2.4.2 Derivation of package limits from repository average limits

It is often the case that the assessment of a particular aspect of post-closure performance is dependent on a repository average value of a parameter. This is particularly true where performance is dependent on the quantity or concentration of a species. In such circumstances, the performance assessment will provide the basis for a limit on the repository average value of the parameter.

The interpretation of a repository limit to provide limits for individual packages is not straightforward. Simply allocating an appropriate fixed fraction of the repository limit to each package neglects the potential variability between packages. In contrast, summing the quantities sent for disposal until the limit is reached may result in the premature closure of the repository if the characteristic is not randomly distributed between packages.

As a compromise, a commonly adopted procedure is to divide the repository limit into smaller limits and allocate each of these on a time and consignor basis. Hence each waste consignor might be allocated an annual limit on a particular characteristic, for C-14 content. Such divisions of the limit need not be equal, and should recognise the expected nature of the waste from the various consignors. The advantage of such a method is that filling of the repository can be monitored and managed without prematurely reaching a limit and forcing closure of the repository. At the same time, the potential impact of the parameter on performance is recognised. This method is, for example, adopted for the management of fissile materials in the UK facility at Drigg.
Where appropriate, the discussion of the derivation of specific criteria indicates that it is actually the repository average value that should be limited. In such cases the above arguments apply.

7.1.2.4.3 Structure of the Discussion

For each of the waste package characteristics listed above, three aspects are discussed. These are the basis of the generic criteria, relevant generic criteria and the methodology for the derivation of specific criteria.
The **basis of the generic criteria** outlines the aspects of repository performance that might be affected by the particular waste characteristic and, in general terms, why this might be the case. This is followed by a listing of **relevant generic criteria** taken from appropriate sources. These criteria are quoted without comment. Finally, the **methods for the derivation of specific criteria** are discussed. This discussion begins with a description of the form that the criteria might take and references to the relevant sections of the waste specific information questionnaire. This is followed by a note of any related waste characteristics. Finally, criteria derivation from operational and post-closures issues are dealt with in turn.

7.1.2.4.4 Radionuclide Content

Basis for generic criterion

- the majority of the risk potentially due to radioactive waste disposal arises from the radionuclide content of the wastes and the demonstration of compliance with many of the remaining criteria is based on calculations, models or arguments that are linked to the radionuclide inventory
- the other criteria may place limits on the total activity, specific activity or concentration of particular radionuclides and may form the basis for acceptance of wastes - hence there is a need for information of a suitable quality

Relevant generic criterion/criteria

- the type and content of radionuclides in the waste package should be known with sufficient accuracy to ensure compliance with authorized limits and should be documented accordingly [7.5]
- the type, characteristics and contents of radionuclides in the waste package should be known with sufficient accuracy to ensure compliance with authorized limits and should be documented accordingly [7.6]
- information on the radionuclide inventory of raw and conditioned wastes is required to ensure that waste packages are produced within specified limits and with known activity contents [7.6]

Methodology to develop specific criterion

- this criterion will comprise a list of the radionuclides for which the quantity present should be reported, together with an indication of the expected accuracy and a lower limit below which the quantity need not be specified
- the radionuclide data supplied by the waste consignor will be compared with the criteria
- the emphasis is on identifying the appropriate quality for the data - the particular radionuclides that should be considered and/or measured for all or particular types of waste
- it should be recognised that it is not practical to require that the quantities of all radionuclides present should be measured and reported (due to short half-lives, difficulty of analysis, small quantities or insignificant potential risk)
- this criterion is <u>not</u> intended to be a list of upper limits on some or all radionuclides (however, such limits may arise from some other criteria)
- the operational safety case and post-closure performance assessment should provide a means of identifying 'key' radionuclides - that is those radionuclides which contribute to risk at a defined level
- should also pay attention to 'fault' scenarios that are not the main basis of the safety case and assessment (for example impact or fire accidents) as these may alter the relative significance of particular radionuclides
- these documents also should provide an indication of the 'quantities' (total activity, specific activity, concentration in solution *etc*) that have a significant effect on risk
- aim to derive values which indicate the minimum quantity of a radionuclide that should be recorded in the waste package data - these may be based on some fraction of the level considered to have an effect on risk
- international experience in establishing such values may be available - this may be dependent on the disposal concept
- need to consider how the radionuclide data should be specified - possibilities include:
  - activity per unit volume or mass of raw waste
  - activity per unit volume or mass of conditioned waste
  - concentration in pore solution for a cemented waste
  - total activity per consignment
- also note that some radionuclides may have different limits depending on the nature of the waste - this is particularly the case for readily dispersible materials such as powders
- the WAC could, in addition, provide a listing of those radionuclides which are judged never to be significant or which are specifically excluded from the repository (for example fissile materials may be excluded)

7.1.2.4.5 External Dose Rate

Basis for generic criterion

- dose from waste packages during the operational phase may adversely affect workers
- dose in the post-closure period not likely to be important in developing WAC (although it may affect processes such as radiolytic gas generation, which are considered later)

Relevant generic criterion/criteria
- the surface dose rate from the waste package should be such that occupational exposures are kept at an acceptable level [7.5]
- external dose rates of conditioned waste packages shall be in compliance with the limits for facilities and equipment in which they will be handled, stored and transported prior to emplacement in a repository [7.6]
- external radiation dose rates of waste packages should be in compliance with the limits established for the facilities and equipment in which they will be handled, stored and transported... [7.7]

Methodology to develop specific criterion
- the criterion will be a limit to the actual dose rate at a specified distance from the package surface (or a set of such limits)
- these data will provide the basis on which the dose rate from a package may be determined for the waste at the time of disposal
- operational phase safety case should include consideration of dose rate during handling
- IAEA Transport Regulations may apply if waste packages have been transported from other sites [7.14]
- may assume some form of regulations will apply to transport on the site even if the waste originates on site (local operating instructions)
- WAC should be a direct repeat of the lowest value from any or all of the above
- if the above do not apply, use the same basis as (for example) IAEA Transport Regulations to calculate for operational phase
- the actual dose rate from a package, and hence the shielding benefits, will depend on the package type and the detailed inventory and therefore will need to be quantified using an appropriate calculation for any given package
- the use of re-usable shielding for transportation and handling at the repository can also be considered

7.1.2.4.6 Surface Contamination

Basis for generic criterion
- contamination of workers and equipment during the operational phase should be limited
- contamination can increase dose (and may not be included in calculations based only on inventory)
- in the post-closure period contamination is not likely to be important in developing WAC

Relevant generic criterion/criteria
- external surface contamination of the waste package should be within limits such that occupational exposures are kept at an acceptable level [7.5]
- transferable radioactive contamination on the exterior of waste packages should be maintained within limits established for storage, transportation and packaging facilities where wastes are to be handled [7.6]
- non-fixed (removable) radioactive contamination on the exterior of waste packages should be maintained within the limits established for interim storage, transportation and disposal [7.7]

Methodology to develop specific criterion

- the criterion will be a limit to the permitted level of non-fixed contamination (α and/or β/γ) on the package surface
- would expect the criterion to specify the area to be sampled, perhaps based on international best practice, for example 300 cm2 from Drigg WAC, or concentrating on the zone most likely to be contaminated.
- operational phase safety case should include consideration of the permitted levels of contamination during package handling
- IAEA Transport Regulations may apply if waste packages have been transported from site to site [7.14]
- assume some form of regulations apply to transport on the site if waste originates on site
- WAC should be a direct repeat of the lowest value from any or all of the above
- if above do not apply, use same basis as (for example) IAEA Transport Regulations to calculate for operational phase

7.1.2.4.7 Structural Stability

Basis for generic criterion

- discussions of generic criteria for structural stability focus on the operational phase and handling requirements
- there is a requirement for packages to be able to withstand impacts during transport and handling (operational phase issues) - impacts should not result in a loss of containment and the subsequent release of activity
- there is also a possible need for packages to be stacked in interim storage and the repository, and hence for the packages to be able to support a sustained load without distortion or loss of containment. For surface disposal facilities this may include the vault slab and earth cap
- both of the above are viewed as primarily properties of the container, although the waste form may provide additional internal support
- the properties of the packages may change during storage prior to disposal (for example due to corrosion of the container) - may need to consider specifying the condition of the package as-received at the repository , rather than as-manufactured
- retrievability may be an issue for some disposal concepts - this may place additional requirements on the maintenance of structure and containment or the need to use an overpack whose integrity has to be guaranteed for a given time period
- post-closure, the package structure will deteriorate due to degradation processes such as corrosion of waste and container and reactions between components of the waste and with external species (for example from groundwater)
- such long term structural stability is also covered under this criterion
- note that generic criteria that include deep disposal may not always place an emphasis on physical containment and hence this may not be covered in the generic criteria [7.5]

Relevant generic criterion/criteria

- the structural stability of the waste form or waste package should be such that occupational exposures are kept at an acceptable level and that the performance of the disposal system is assured [7.5]
- the waste packages must be able to withstand stresses without unacceptable deterioration in their ability to accomplish safety related functions [7.6]
- waste packages shall be designed such that, in conjunction with handling systems, releases due to mechanical impact under foreseeable incidents are limited to acceptable levels [7.6]
- the properties of the container, including (where applicable) mechanical properties...(and) durability...should be controlled to ensure they comply with requirements for safe handling, storage and disposal ... [7.7]
- the waste package must be able to withstand mechanical stresses arising during handling, transport, storage and emplacement in a repository under both normal and predictable abnormal situations [7.7]

Methodology to develop specific criterion

- it is not straightforward to determine what would be specified in this criterion, as 'structural stability' is not a simple, measurable quantity
- the criterion could encompass several different properties, depending on the detail of the safety cases and performance assessment, including the following items:
  - mechanical strength of the container
  - mechanical strength of the waste form
  - response of the container or waste form to tensile stresses
  - dimensional stability (retention of original shape, minimisation of expansion)
  - conditioning of the waste to minimise changes in the post closure period
- the criteria would be a listing of minimum requirements for some or all of the above (and possibly other issues)
- note links to chemical durability, corrosion resistance and gas generation
- the development of the specific criteria might need to consider three different areas:
  - mechanical properties not related to radionuclide release
  - radionuclide release in the operational phase
  - compatibility with post-closure requirements
- 'mechanical' requirements would be expected to be developed from more detailed repository designs
- this area encompasses issues such as stacking loads and the maintenance of handling features - an assessment should give requirements for simple mechanical properties such as the overall strength of the package
- for criteria based on radionuclide release in the operational phase, need to consider possible outcomes and work backwards
- the following scenarios might be considered:
  - release of waste material in the operational phase due to impact damage
  - release of waste material in the operational phase due to fire damage
  - degradation of a package during storage or before closure, compromising containment in the operational phase (radionuclide release into an aqueous phase in contact with the packages)
  - release of gaseous radionuclides in the operational phase (for example radon and H-3)
- the waste packages should be designed/specified such that the limits will not be exceeded, and the criteria will reflect this need

- scenario(s) will determine how a release may occur and link this to package properties through engineering models or experience - for example, assume a specified impact such as a drop from 10 m then determine expected release and relate to package properties
- this could be covered by a requirement for waste consignors to provide the demonstration of performance against a specific set of events (for example a range of drop heights)
- the post-closure performance assessment will make some assumptions about the properties of the waste and containers (this might include a 'null' assumption that such properties are not considered)
- the criteria may need to reflect these assumptions
- most significant is the concept of containment - a period during which radionuclides are retained in the packages
- a simple approach might be to specify a time period before radionuclide release into groundwater takes place (a package lifetime) - this can have practical difficulties, for example if the loss of containment is assumed to be due to corrosion, what extent of corrosion (fraction of the container penetrated) constitutes loss of containment?
- stability of the waste form may be an issue - processes such as expansive corrosion and gas generation can breach containment and hence these should be minimised - again these would need to be linked to a quantitative criterion
- could require an assessment of the evolution of the waste package under post-closure conditions to demonstrate the retention of containment
- requirement for 'best practice' is not really a specific criterion, although this might be the only useful method

7.1.2.4.8 Chemical Durability

Basis for generic criterion
- there is a general requirement for the waste package to maintain a particular set of conditions or properties when exposed to an 'agent of change' - this is mainly to maintain consistency with the assessed condition
- do not expect to base a performance assessment primarily on chemical behaviour or conditions (that is low radionuclide solubility) as shallow burial is based primarily on the containment provided by physical barriers - the main issue is to ensure that the containment is not compromised
- this is mainly a longer-term issue, although waste-matrix interactions could come in here - although this might be regarded as a structural stability issue
- would not expect to see significant deterioration during storage

Relevant generic criterion/criteria
- the chemical durability of the waste package should be sufficient to provide the required containment of radionuclides in the disposal environment [7.6]
- the chemical durability of the waste package should be sufficient to provide the required containment of radionuclides during interim storage, transport and disposal [7.7]

Methodology to develop specific criterion

- the criteria would be based on the requirement that the waste package should not respond to the chemical conditions expected in the repository in such a manner that the containment of radionuclides is compromised
- the specific criteria would need to refine this definition through the description of the repository environment that has to be endured and the more detailed specification of the containment requirements (for example radionuclide release fractions as a function of time)
- would expect to arrive at a range of particular requirements which will be specific to the waste, package and disposal concept
- note links to corrosion resistance
- would expect to develop requirements from scenarios used in safety cases or performance assessments, for example the maintenance of containment for a specified period when exposed to a particular groundwater
- for shallow burial, all arguments are likely to be 'secondary' - chemical durability is required to ensure that structural stability is maintained, which in turn provides consistency with the safety case or performance assessment
- would need to consider how chemistry can impact on performance through appropriate mechanisms/scenarios, which would be specific to the waste, container and matrix - examples might include sulphate attack on cements, the effect of acidic groundwaters, gas generation, corrosion or degradation of waste and waste-matrix interactions
- in general, this characteristic refers primarily to the waste form
- the mechanisms are generally dependent on several parameters and it is not possible to provide chemical durability through the simple limitation of a single component (hence corrosion is influenced by parameters such as pH, temperature, chloride concentration, reactive metals and the presence of corrosive liquids)
- the probable evolution of a proposed waste package should be considered in relation to a chemical environment appropriate to the disposal concept - this will demonstrate whether the waste is consistent with assumption regarding containment
- all processes that might lead to loss of containment might be examined to indicate what limits might be set - for example, the groundwater chemistry might be such that only stainless steel would resist corrosion to the required extent
- this could lead to limits on particular species or materials, or a requirement to use particular materials (for example bitumen, sulphate-resistant cement or stainless steel) - but such limits cannot be generalised in advance

7.1.2.4.9 Corrosion Resistance

Basis for generic criterion

- assume this characteristic is referring to the packaging/container
- primary concern is the maintenance of handling/containment during the interim storage period and operational phase (assuming packages are predominantly metallic), which will place a limit on the extent of corrosion
- physical containment in post-closure period also may place limits on the extent or rate of corrosion of the container
- waste corrosion is dealt with under chemical durability

Relevant generic criterion/criteria
- containers should be designed with sufficient corrosion resistance to meet the requirement for their normal life and intended use [7.5]
- the properties of the container, including (where applicable)...durability...should be controlled to ensure they comply with requirements for safe handling, storage and disposal... [7.7]

Methodology to develop specific criterion
- main issues are how can 'corrosion resistance' be specified (that is what actual parameters would be used) and how can appropriate values be obtained
- ultimately, the criterion could take the form of a list of permitted materials (this links to the idea of standard packages) or a description of the permitted behaviour (fraction of surface penetrated in a specified time, corrosion rate, container thickness etc)
- could describe the assumed behaviour (and/or material) and require the consignor to demonstrate that their proposal would give an equivalent performance
- note links to structural stability and gas generation
- the packages should be able to provide the necessary mechanical strength subsequent to any corrosion during the storage and operational phases
- basis will be the assumption made in the post-closure performance assessment about the contribution of the container to containing radionuclides (if no such assumption is made, there will be no WAC from post-closure performance)
- need to develop an argument that links the extent of corrosion of the containers to the radionuclide source term - this could be complex or simple and will be covered in the performance assessment
- need to develop a criterion that ensures that the containers will be consistent with the assumptions made in the assessment
- also need data on corrosion behaviour of materials to ensure that the models are realistic - data would include corrosion rates (and dependence on environment), nature of corrosion (local, general, crevice, etc.), variation with time and some information on variability between packages
- operational safety case will also include some (perhaps implicit) assumptions about the stability of the package - this could be covered in 'structural stability'

7.1.2.4.10 Thermal and Radiation Stability

Basis for generic criterion
- need to demonstrate that heat and radiation do not degrade the performance of the package (such that the assumptions in operational or post-closure cases are invalid)
- could also extend to include fire accident performance
- note that the cumulative dose in a shallow repository is likely to be quite small as most species are relatively short-lived

Relevant generic criterion/criteria
- waste packages whose thermal or radiation energy outputs would jeopardize the performance of the disposal system should not be accepted for disposal [7.5]
- the thermal power output of the waste packages shall comply with limits applicable for storage, transportation and handling prior to emplacement [7.6]
- the thermal power output of the waste forms shall be limited such that any associated changes to physical, chemical and mechanical properties of the waste form, waste

package components, other engineered barriers and repository components and the host geology do not adversely affect the safety of the overall disposal system [7.7]
- waste packages shall be designed so that, in conjunction with handling systems, releases due to thermal impact under foreseeable incidents are limited to acceptable values [7.6]
- radiation dose rates shall be controlled to levels sufficient to ensure that radiation-induced processes (such as radiolysis) and degradation of material properties of the waste packages, repository components and the host geology do not occur to an unacceptable degree [7.6]
- short and long term influence of radiation (and) heat...on the waste package's integrity must be assessed [7.7]
- the fire resistance of waste packages should be demonstrated to comply with all applicable specified requirements, safety assessments and codes covering interim storage, transportation and disposal [7.7]

Methodology to develop specific criterion
- need to quantify the heat load and radiation dose expected in the repository, assess the likely effects and use this to devise limits for packages (such that the assumptions are not invalidated)
- criterion would specify the heat load and dose (perhaps in the form of radionuclide content) and list any properties judged to be potentially affected - consignor might need to demonstrate that the package would not be adversely affected
- alternative might be to develop an argument based on standard packages which are capable of meeting the requirements - use of such packages would then remove the need to set specific WAC in this area
- heat comes from radioactive decay, chemical processes (cement hydration, corrosion and other reactions) and microbial action
- suggested approach is to produce a model of the repository and estimate temperature rise (repository design may be modified to ensure temperature is acceptable) - this will need assumptions about thermal properties, waste inventory etc.
- the potential effects of the temperature rise could then be considered and shown to be acceptable - possible effects might be acceleration of chemical processes such as corrosion, changes in materials, effects on microbial action and water loss/drying
- if the effects are found to be small or not significant, the assumed values of the properties needed in the assessment could be relaxed
- the criterion would then be that the properties of the waste should not deviate significantly from the assumed values (the issue of criteria based on repository average properties, and acceptable deviations from the average, might need to be dealt with elsewhere)
- the radiation dose can be similarly estimated for the average dose rate and the potential effects considered (may be able to concentrate on a few key radionuclides) - might include effect on corrosion and gas generation
- need to use a suitable average - should be based on a well-founded inventory
- the radionuclide inventory will show significant variations between packages and hence the issue of criteria based on repository average properties will be pertinent
- would need to develop a permitted range of inventories for some radionuclides, but exceptions might have to be allowed

7.1.2.4.11 Combustibility

Basis for generic criterion

- fires during the operational phase present an increased risk of radionuclide release and hence precautions to minimise the risk of fire are required
- require appropriate steps to minimise the risk of combustion

Relevant generic criterion/criteria

- the combustibility of the waste form or waste package should be such that the potential for fire is as low as reasonably achievable [7.5]
- intermediate and low level alpha waste packages containing combustible materials may be acceptable if the containers are non-combustible or heat resistant and do not support combustion under foreseeable incidents [7.6]
- the fire resistance of waste packages should be demonstrated to comply with all applicable specified requirements, safety assessments and codes covering interim storage, transportation and disposal [7.7]

Methodology to develop specific criterion

- criterion might be the requirement to exclude or limit the quantities of combustible materials in a waste package - this may require an appropriate definition of combustible materials or a listing of materials to be considered
- an alternative criterion might be the requirement for an assessment of the combustibility of the waste and a demonstration that the conditioning and/or packaging of the waste will render any potentially combustible components of the waste effectively non-combustible
- these two criteria may both apply - establish limits for specified combustible materials above which a specific assessment and demonstration of safety is required
- need to consider the conditions under which combustion should be considered possible (ignition source, external temperature, oxygen availability)
- in combination with a waste inventory and general knowledge of likely waste compositions, these conditions should provide a list of combustible materials that might be present in wastes to be drawn up - this should provide guidance to the waste consignors as to the materials to be considered, although such a list should not be regarded as comprehensive
- if limits are to be established, consideration should be given to the potential effects of combustion - for example, the additional heat load due to complete combustion may be compared with the heat capacity of the conditioned waste to determine a maximum possible temperature rise, which in turn may be used to establish a quantity of combustible material for which the temperature is judged to be not significant
- the operator may wish to provide an illustration of a combustibility assessment
- this might include the description of standard waste conditioning processes that would be accepted as rendering waste non-combustible or at least increasing the limits on the quantities of combustible materials - for example, the supercompaction of waste might be judged to prevent the combustion of plastic and paper wastes

7.1.2.4.12 Gas Generation

Basis for generic criterion

- gas generation may result in pressurisation of a waste package, threatening the integrity of the package and containment
- loss of containment during the operational phase may increase risk and dose to workers through the release of radionuclides
- the generation and release of flammable gases (such as hydrogen), toxic gases (such as hydrogen sulphide) and radiotoxic gases (such as radon) during the operational phase may present an increased risk to workers either directly or indirectly
- during the post-closure period, loss of containment due to over pressurisation may invalidate the assumptions upon which the performance assessment is based
- in addition, gas generation may disrupt or alter groundwater flow compared with that assumed in the performance assessment
- the potential release of gases generated within a repository to the human environment should be considered as part of a performance assessment - this might include both radiotoxic and non-radiological effects (such as explosions, fire and toxicity)

Relevant generic criterion/criteria

- gas generation in the waste package should be such that the performance of the disposal system is not compromised [7.5]
- gas generation in the waste package or in surrounding media should not jeopardize the performance of the overall disposal system [7.6]

Methodology to develop specific criterion

- a simple criterion would be a limitation on the permitted rate of gas generation from a waste package
- this might be qualified by reference to the properties of the waste form and package such that the gas generation rate is limited to a value that will not cause unacceptable pressurisation and threaten containment
- when considered for operational phase safety, the criterion is likely to be derived from, and refer to, the behaviour of individual packages
- when considered for post-closure performance, the criterion is more likely to be referred to a repository average value (for example, groundwater flow may be most affected by the total gas flow from the repository)
- limits may be placed on the generation of specific gases which have a direct impact on risk, for example radiotoxic or toxic gases (such as radon, H-3 and hydrogen sulphide)
- note links to structural stability, chemical durability, corrosion resistance and microbial degradation
- gas generation may compromise the integrity of a waste form through pressurisation
- the pressure rise may be predicted through a model of gas migration in the waste form, taking account of the properties of the waste form (permeability), the generation rate and the geometry of the system
- an unacceptable pressure may be determined through comparison with the tensile strength
- gas generation may compromise the integrity of a waste container through pressurisation
- the pressure rise may be predicted by comparing the gas generation rate with the rate of release of the gas from the container through a vent or other leakage

- an unacceptable pressure may be determined through comparison with the tensile *strength of the container material*
- this may also demonstrate the need for venting of the package
- consideration of the potential effect of gas in the post-closure period may be based upon the assumptions made in the performance assessment (should this take account of gas generation)
- limits may be established if groundwater flow is affected to the extent that the risk is increased unacceptably
- for a shallow repository, pressurisation and loss of repository integrity should be considered
- it is noted that the above assessments may establish limits for the repository rather than individual packages
- the risk arising from the generation of toxic, radiotoxic and flammable gases should be considered separately from any risk arising from a loss of containment due to pressurisation
- the release rates for potential toxic, radiotoxic and flammable gases should be established and compared with the operational phase safety case
- consideration of such gases may place limits on specific components of the waste which give rise to these gases
- a number of mechanisms should be considered in the assessment of gas generation rates - these may include:
  - anaerobic corrosion of steels
  - corrosion of other metals such as aluminium and zinc
  - microbial degradation of organic materials
  - radiolysis of pore solutions and organic/putrescible materials
  - generation of radioactive gases through radioactive decay (e.g. radon)
- some/all mechanisms will depend on the waste conditioning, as the chemical parameters such as pH and Eh can be important
- input parameters for estimates may be based on experimental data provided by waste consignors or it may be beneficial to provide a standardised method for estimating gas generation rates from wastes, based on available corrosion rates *etc*, for the use of consignors

7.1.2.4.13 Microbial Degradation

Basis for generic criterion
- the microbial degradation of organic materials in the waste and the growth of a microbial population may have an effect on the performance of the disposal system
- the potential effects include gas generation, the formation or consumption of complexants, the generation of acidic waste products, microbial-induced corrosion (MIC)
- these effects may have an adverse impact on the actual post-closure performance compared with that predicted in an assessment

Relevant generic criterion/criteria
- in order to control microbial degradation, the presence of organic material should be limited so that the performance of the disposal system is not compromised [7.5]

Methodology to develop specific criterion

- the criterion may impose an exclusion or limitation on particular components of waste that promote microbial growth, or which may be anticipated to have a adverse effect on performance when subject to microbial degradation
- note links to **gas generation**
- the development of specific criteria will require the identification of materials of concern and the quantification of appropriate limits
- it should be recognised that microbial degradation in the post-closure environment is not totally predictable and will depend on the size, nature and distribution of the microbial population either originally present in the repository or introduced at a later stage, as well as depending on the environmental conditions and their variation with time and location
- it is therefore appropriate to make conservative assumptions regarding the extent and nature of the microbial degradation processes - any particular organic material should be assumed to be degraded to give the maximum possible yield of any species that has an adverse effect on performance
- a post-closure performance assessment may take account of the effects of microbial degradation - the following processes may be considered:
  - generation of gases, especially carbon dioxide and methane
  - generation of toxic gases such as hydrogen sulphide
  - modification of the quantities of chemical complexants
- these processes may be represented as 'microbially-catalysed chemical reactions' which can be described in models using appropriate rate constants - an example is the gas generation model GAMMON, developed for UK Nirex [7.5]
- models can be used to investigate the potential effect of different inventories of organic materials on post-closure performance and, based on the resulting predictions, establish limits for the quantities of some organic materials if appropriate
- the distribution of the microbial population within the repository will be unpredictable
- hence microbial degradation is commonly dealt with on the scale of the whole repository, based on the total inventory of the appropriate organic material
- this will result in the derivation of repository average limits, which would have to be interpreted to give package limits
- it is recognised that the prediction of the evolution of the microbial population in detail is not possible and, with appropriate arguments, the modelling of the effects of microbial degradation may be omitted from an assessment - this is particularly the case when the requirement is for relatively short-term containment
- in such a case it is appropriate to attempt to exclude materials that would obviously promote rapid microbial growth - hence the common exclusion of putrescible materials
- exclusion could be extended to other organic materials if these could be shown to promote rapid microbial growth
- the difficulty of demonstrating complete exclusion is noted

7.1.2.4.14 Presence of Free Liquids

Basis for generic criterion

- free liquids represent a mobile phase that may be more readily released under abnormal and accident conditions during the operational phase - such liquids may contain activity
- at all stages free liquids may promote potentially adverse chemical processes such as corrosion and microbial degradation - pockets of liquids within a waste form may

provide environments isolated from the conditioning matrix where differing chemical conditions may persist, negating the benefit of the conditioning
- liquids sorbed onto other material also may be included
- the voidage associated with free liquids may affect structural stability

Relevant generic criterion/criteria
- the amount of free liquids within waste packages should be sufficiently low to ensure that occupational exposures are kept at an acceptable level and that the performance of the disposal system is not jeopardized [7.5]
- the quantity of free liquids in waste packages should be sufficiently low to ensure that the performance of the overall disposal system is not jeopardized [7.6]
- materials that promote migration of released radionuclides or hazardous substances (free liquids)... should be prohibited in the waste package [7.7]

Methodology to develop specific criterion
- simple criterion would be to exclude all free liquids
- difficulty arises in demonstrating that a waste package does not contain any free liquids - hence a limit may be adopted
- this may require that a conditioning process is adopted that is specifically designed to minimise free liquids (for example drying or cementation)
- note links to corrosion resistance
- there is some difficulty in establishing a quantitative limit on free liquids - mainly because the magnitude of the effect of free liquids may not be directly linked to the quantity of liquid but the type and location
- in addition, the distribution of the liquid in the conditioned waste and the nature of the waste also may be important
- reference to the mobilisation of activity in an accident condition provides an alternative means of setting a limit, provided that the activity in solution can be quantified and related to the volume of liquid
- the volume of free liquid could be limited to the volume which would contain the maximum activity that is permitted to be released
- this could be assessed by reference to the expected radionuclide content of wastes
- alternatively, the volume could be limited to that which is readily detectable in a waste form
- there is no benefit in setting a limit that is below the limit of detection
- in this case, the criterion would be 'no detectable voidage' in the waste form
- minimisation of free liquids may be seen as an appropriate goal

7.1.2.4.15 Presence of Chelating or Complexing Agents

Basis for generic criterion
- experimental work has shown that, for some repository concepts, the solution concentration of some important radionuclides is increased in the presence of particular chelating or complexing agents - these potentially may compromise the assumptions made in post-closure performance assessments or may have been assessed as giving rise to unacceptable risk
- the sorption of some key radionuclides may also be adversely affected, compromising assumptions made regarding the contribution of sorption to containment

- it is noted that this characteristic is not likely to be a major concern for the operational phase safety case and the potential impact on post-closure performance is dependent on the repository concept

Relevant generic criterion/criteria
- waste forms that contain chelating or complexing agents should be treated or packaged to prevent the enhancement of radionuclide migration [7.5]

Methodology to develop specific criterion
- the criteria may comprise limits on specific compounds that are known to enhance radionuclide solubility (or reduce sorption) and (perhaps) limits on compounds or materials that may be degraded or altered under repository conditions to yield products that enhance solubility (or reduce sorption)
- the absolute exclusion of certain compounds may be difficult to demonstrate and therefore providing limits is preferable - furthermore, it may be unrealistic to exclude some potential complexing species, for example cellulose degradation products, if they arise from wastes
- there are a very large number of possible complexing species and it is not realistic to attempt to set limits on all such species
- in practice, criteria probably will be limited to those complexing species that are judged likely to be present, based on some prior knowledge of the nature of the waste
- the post-closure performance assessment will have identified the key radionuclides and will have been based upon assumptions regarding the solubilities (and sorption behaviour) of those radionuclides
- possible complexing agents in the waste primarily should be identified through a waste inventory
- the identification of possible complexing agents arising from the degradation of waste components will depend on both the nature of the waste and consideration of potential degradation
- the performance assessment may have considered the effects of complexing agents, in which case the appropriate limits on the quantities of such materials may be based on the assumed concentrations which have been shown to have no significant effect on risk
- in the absence of such an assessment, the assumptions in the performance assessment should be reviewed in the light of published experimental data, other performance assessments and general scientific reasoning to determine whether significant solubility increases are to be expected and to provide information regarding the potentially acceptable levels of complexants
- solubility enhancement is primarily a concern for actinides, and it may be possible to demonstrate that the anticipated levels of such species are sufficiently small that any reasonable solubility increase would not greatly affect risk
- this also might be the case if the post-closure performance assessment is based solely or largely on the physical containment of short-lived species - solubility might only be a concern for 'fault' scenarios where containment is reduced or lost
- it is noted that performance assessments may be based on repository average data and as such will provide limits on the average quantities of complexing species
- the issue of establishing package limits based on repository average limits may be applicable in this case

- more detailed assessments may be required to establish the potential significance of localised concentrations of complexants as it is not necessarily the case that performance is a linear function of quantity of complexant
- may need to consider how a consignor can demonstrate compliance - this may require an argument based on the source and nature of the waste or a demonstration that the waste treatment/conditioning process removes or changes any potential complexing species originally present in the waste
- where significant materials are known to be present, the levels may be measured (for example with cellulose)
- the timescale also should be taken into consideration - some complexing species may be formed slowly by the degradation of other materials (for example cellulose) and therefore might not be a concern for a repository for predominantly short-lived wastes

7.1.2.4.16 Explosivity and Presence of Compressed Gases

Basis for generic criterion
- explosions and the explosive dispersal of activity during the operational phase potentially will result in increase doses to workers and may cause damage to equipment and other packages
- the actual significance of explosions subsequent to closure may depend on the disposal concept, but generally such events should be regarded as undesirable

Relevant generic criterion/criteria
- waste packages that contain explosive materials should not be accepted for disposal [7.5]
- waste packages containing materials that might cause explosion...hazards shall not be accepted for disposal [7.6]
- containers of compressed gases should not be accepted for disposal [7.6]
- items that could pose safety concerns (containers and pressure vessels within the waste package)... should be prohibited in the waste package [7.7]

Methodology to develop specific criterion
- the specific criterion may resemble the generic criteria cited above; the exclusion of potentially explosive materials including compressed gases
- explosive materials may require further definition in a criterion
- in practice, it may be difficult to demonstrate the exclusion of explosive materials except through arguments based upon the source and nature of the waste and the corresponding treatment/conditioning etc.
- further complications may arise if combinations of materials are potentially explosive (an example of this is the production of hydrogen gas and subsequent mixing with oxygen or air during the operational phase) and a consignor may be required to give appropriate consideration to such possibilities
- it may be necessary to consider setting limits on some materials, rather than requiring complete exclusion
- arguments also may be based upon changed chemistry due to the conditioning process
- it is likely that the exclusion of compressed gases may be demonstrated experimentally through appropriate assay of the packaged waste (for example through x-ray imaging)

7.1.2.4.17 Pyrophoricity

Basis for generic criterion

- fires during the operational phase present an increased risk of radionuclide release and hence precautions to minimise the risk of fire are required
- removal of potential sources of ignition is therefore desirable
- given that most conditioned waste is unlikely to be combustible, ignition of flammable gases is perhaps the main cause for concern

Relevant generic criterion/criteria

- prior to disposal, waste forms that contain pyrophoric materials should be treated or conditioned to eliminate hazard due to this characteristic [7.5]
- waste packages containing materials that might cause ignition hazards shall not be accepted for disposal [7.6]
- waste form with properties that could contribute to risk (pyrophoricity) should be prohibited in the waste package [7.7]

Methodology to develop specific criterion

- need to identify pyrophoric materials that might be in the waste and suggest means for removing the risk (pre-conditioning the waste) - possible examples included reactive or finely-divided metals
- ideal would be to exclude such materials (requirement to modify to a safe condition is assumed to be the same as exclusion)
- the minimum quantities which might give rise to risk could be identified - recognising that complete exclusion cannot be demonstrated

7.1.2.4.18 Toxicity and Corrosivity

Basis for generic criterion

- radioactive waste may or will contain materials that pose a risk to health due to their toxicity (for example lead, cadmium, mercury *etc*) and which may therefore need to be limited
- regulatory bodies may require a demonstration that the waste conditioning also will minimise the risk from these materials (it is possible that the toxic risk could be greater than the radiotoxic risk, especially for SLLW)
- wastes also may contain materials that are aggressive towards metal containers, other (different) wastes and waste forms and the surrounding repository components (the term corrosive materials is adopted, although reactions other than corrosion also may be promoted) - maintenance of the desired performance of the packages and repository may require such materials to be limited

Relevant generic criterion/criteria

- the contents of non-radioactive toxic and hazardous materials should be known with sufficient accuracy to ensure compliance with authorised limits [7.6]
- waste forms should not contain materials which will corrode the waste containers or other barriers in the disposal system [7.6]
- ... waste form with properties that could contribute to risk (...corrosivity, reactivity...) should be prohibited in the waste package [7.7]

Methodology to develop specific criterion

- the criteria will comprise a list of particular materials and limits on the quantities allowed in the waste
- note links to corrosion resistance
- the potential risk from toxic materials may be assessed using a methodology similar to that used for radioactive releases (development of scenarios, assessment of risk *etc*), indeed the scenarios would be expected to be the same
- the wastes should be sufficiently well characterised to allow a detailed assessment for all possible toxic materials - also there are a large number of potentially toxic species
- an alternative method might be to identify representative species (for example dissolved toxic metals, organic species etc) and assess these
- toxic materials in groundwater/drinking water are likely to be regulated by numerous national or international bodies (for example 'red list' materials in the UK) - these limits may be linked back to repository contents through information on solubility under repository conditions to derive repository limits on quantities
- the issue of establishing package limits based on repository average limits may be applicable in this case
- it may be possible to identify corrosive materials from general experience (for example acids or species that may degrade to yield acids) but it is less straightforward to derive limits for these species
- the basis of an argument will be the requirement to be consistent with the assumptions made in a performance assessment
- would need to restrict the quantities of corrosive materials such that, for example, the rate of container degradation was not unacceptably accelerated compared with that assumed in a performance assessment - however, it may be difficult to make the quantitative link between the quantity of the corrosive material and the rate of degradation or the impact of the degradation
- could require the consignor to demonstrate that all potentially corrosive materials have been rendered inert by the waste conditioning process - this would be in the form of an argument or experimental demonstration
- <u>but</u> this requires the recognition that corrosive materials are present
- it is noted that the detailed assay of non-radioactive materials is more difficult than for radioactive materials (remote measurement is not possible, and analysis of materials may be hindered by activity content)
- hence it may not be practical to require the exclusion of such materials, as it is not possible to demonstrate that this has been achieved
- guidance on the derivation of limiting values for chemical toxic components in waste packages has been addressed in a recent CEC report [7.15]

7.1.2.4.19 Criticality Safety

Basis for generic criterion

- fissile materials may be present in some wastes in quantities sufficient, under appropriate circumstances, to give rise to criticality concerns (either within a single package, in an array of packages or through re-location and accumulation of material in the repository environment)
- a criticality incident during the operational phase might result in excessive dose to workers and therefore is considered unacceptable

- a criticality incident is generally considered to be undesirable as it would result in additional heat loading and would modify the radionuclide inventory compared with that assumed in assessments

Relevant generic criterion/criteria
- the content of any fissile materials in waste packages should be limited so that critical conditions will not result [7.5]
- waste packages shall be designed to preclude nuclear criticality of a single waste package. Handling, storage and disposal systems shall be designed and operated to ensure that criticality of arrays of waste packages cannot occur [7.6]
- the fissile content of wastes should be controlled to ensure that sub-critical conditions are maintained under all conditions likely to be encountered at any time during conditioning and to ensure that the requirements of the waste package specification are met [7.7]

Methodology to develop specific criterion
- the criterion may be the exclusion of defined fissile materials (or the statement of limits which are judged to be effectively the same as exclusion) or a more detailed description of the allowed limits and the circumstances under which they may be applied
- appropriate criticality assessments also may be required, especially when the quantities of fissile materials are significant
- fissile radionuclides should be clearly identified
- fissile materials may not be present in the waste and therefore not of concern
- fissile materials may be 'excluded' from the repository, provided that this is consistent with the expected nature of the waste (this may be a regulatory requirement)
- however, it is necessary to consider the definition of exclusion - this will depend on the origins of the waste, the assay methods available and the quality of information on the waste
- in practice, an upper limit on the quantities of defined fissile materials might be more appropriate - this could be based on a criticality assessment for the repository (or packages) or on data recording issues
- the issue of establishing package limits based on repository average limits may be applicable in this case
- where fissile materials are to be accepted, the limits should be derived from appropriate criticality safety assessments
- several types of assessment may be performed:
  - short-term package assessment (the packages as-manufactured)
  - long-term package assessment (the packages as-evolved under repository conditions)
  - overall repository case (consideration of all packages and the migration of material)
- such cases may be used to establish limits for individual fissile materials (U-233, U-235, Pu-239 and Pu-241 are usually considered)
- it may be possible to derive a generic limit for the fissile materials which is judged to be safe under all foreseeable circumstances - this would have to take into account the likely nature and variability of the waste
- this generic limit could be applied as a WAC, or used as 'break-point' to define the point at which the detailed assessment of a specific waste package will be required
- consideration of accuracy and variability between packages of the same waste type is important (also the ability of the waste conditioning to produce a package consistent

with idealised description on which an assessment would be based) - further demonstration of the compliance of the conditioning with the assessed scenarios would be necessary
- assessments may be based on suitable scenarios which would be consistent with the general understanding of the repository concept

7.1.2.4.20 Criteria not related to waste package performance

7.1.2.4.20.1 Physical dimensions and weights

Basis for generic criterion
- The description, dimensions and weight of the container should be controlled to ensure they comply with requirements for safe handling, storage and disposal as detailed in the waste package specifications. For example, the dimensions and weight should be such as to be compatible with all handling stages, e.g. lifting equipment, stacking arrangements and cell access

Relevant generic criterion/criteria
- waste packages should be standardised in such a manner that they are compatible with handling and disposal procedures [7.5]
- the physical dimensions and weights of waste packages should be compatible with provisions for transport, handling and emplacement [7.6]

Methodology to develop specific criterion
- The normal approach to this will be to specify the types of acceptable standard containers. In most cases the WAC will also include a caveat such that other containers can be considered on a case by case basis with prior approval from the repository operator
- Manufacture of the container should be carried out within a QA programme in accordance with documentation fabrication procedures and inspection schedule
- Conformance of the container to the required specification may be verified by carrying our independent surveillance and inspection of the manufacturing process

7.1.2.4.20.2 Unique identification

Basis for generic criterion
- Package identification is necessary to ensure safe handling, appropriate emplacement, accountability, radioactivity control, criticality control, inventory control and certification of compliance with specifications

Relevant generic criterion/criteria
- waste packages should be easily identifiable [7.5]
- each waste package for emplacement in a repository should be marked with a unique identification [7.6]

Methodology to develop specific criterion
- The requirement for labelling should specify the label specifications with respect to location on the package/container, size, colour, etc, and the required durability of the label. The time period for which the label should be clearly legible will depend on the

interim storage period, operating regime of the repository. It may typically specify a time period to closure of the particular cell or final closure of the whole repository

7.1.2.4.20.3 Responsibilities and organisation

Basis for generic criterion

- A fundamental requirement of any safe waste management system is that the organisation and responsibilities are clearly defined

Relevant generic criterion/criteria

- Before acceptance of the waste for disposal, a programme of quality assurance should be established and responsibilities assigned, including responsibility for the quality of the waste and waste packages [7.6]

Methodology to develop specific criterion

- The organisations responsible for quality control of the waste should demonstrate and record that the quantitative waste acceptance criteria are being followed
- The quality of waste packages received for disposal should be checked by the radwaste management agency, repository operator to ensure compliance with the waste acceptance criteria. This control may be partially performed at the repository, at the storage facility of the waste producer, and possibly also by an independent control institute

7.1.2.4.20.4 Control of conditioning process

Basis for generic criterion

- Quality control of waste packages is crucial to ensure compliance with waste acceptance criteria. The two important aspects to quality control in this context are the control of the conditioning process and checks on waste packages

Relevant generic criterion/criteria

- Control of the conditioning process should assure the required quality of the waste form and other parts of the waste packages [7.6]

Methodology to develop specific criterion

- As part of the process of securing an authorisation for disposal, it is normal practice for the repository operator to require a submission including details of the conditioning process for a given waste stream. This submission will demonstrate that the conditioning process is able to produce waste packages of the required quality i.e. to the requirements of the waste package specification.

7.1.2.4.20.5 Checks on waste packages

Relevant generic criterion/criteria

- Checks on waste packages should be made to assure the quality of the waste from and other parts of the waste package [7.6]

Methodology to develop specific criterion

- The relevant properties of the waste packages should be checked with non-destructive or destructive test methods. Where ever possible, non-destructive methods should be used to minimise the generation of secondary wastes and the requirements for repackaging of wastes. This checking procedure could be performed at the repository, at the storage facility of the waste producer, and possibly also by an independent control institute.

7.1.2.4.20.6 Disposal record and data recording

Relevant generic criterion/criteria

- The results of the quality control should be recorded [7.6]
- Those record essential to demonstrating conformance of the waste package to requirements contained in the waste package specification should be identified and both the format and information to be provided for individual packages defined [7.7]

Methodology to develop specific criterion

- All the relevant data of the waste producer, conditioner and repository operator should be recorded. The record system of the parties involved should be compatible to allow information exchange. It should be the responsibility of the competent national authority to specify for what period of time the records should be retained.

7.1.2.4.20.7 Compliance with codes and practices

Relevant generic criterion/criteria

- Waste forms and waste packages shall be certified as being in compliance with applicable codes and standards prior to acceptance for disposal

Methodology to develop specific criterion

- Applicable codes and standards will be as determined by the competent national authorities

7.1.2.4.20.8 Disposal facility management

Criteria may also be included to control the volumes of particular waste types consigned to the repository and to ensure that the utilisation of available repository capacity is optimised.

Examples of such criteria are:

- That the waste consignor shall ensure that waste should not be consigned to the repository if reasonably practicable measures could be adopted to segregate it and consign it to a lower category of disposal facility, e.g. domestic refuse facility;
- When possible, all compactable waste should be supercompacted;
- Limits may be imposed on the amount of decommissioning waste that can be consigned from each waste consignor per year to ensure capacity for future operational waste is maintained.

7.1.3 Waste characterisation methods

For characterizing radioactive waste, two important categories of methods can be applied to define their critical nuclide inventories:

- the first category consists of measurement methods, which are described in more details in Section 7.4 of the present chapter;
- the second category consists of predictive calculation models, which can be validated on the basis of the aforementioned measurements methods.

Section 7.1.3 focuses on such predictive calculation models, on the basis of the example of the LLWAA (Low Level Waste Activity Assessment) and LLWAA-DECOM (Decommissioning) codes [7.16-7.17] that are used to characterize waste generated by Belgian NPPs.

7.1.3.1 Introduction

In order to determine the critical nuclides inventories, different approaches exist and the methodologies currently in use in most EU-countries are the following:

- Simplified calculation approach based on engineering judgement and extrapolation of international experience in similar types of LWR;
- Selection of ratios between the activity inventory of an easy to measure key nuclide and the activity inventory of a critical nuclide (or spectra), specific to each type of waste and each type of plant (BWR, PWR), from experimental databases;
- Predictive calculation codes taking into account the plant's characteristics and operating conditions.

Belgium uses the LLWAA code [7.16] that enables to predict the global inventories and/or the scaling factors of the critical nuclides in the conditioned and in the non-conditioned waste generated by the operation of a nuclear power plant. Based on a required input data sheet, the LLWAA code model can be tailored to the specific characteristics of a given plant.

One significant benefit of the code results from it enabling to easily perform sensitivity analyses to assess the effect on the nuclide inventories due e.g. to the change of some materials, of operating procedures or operating conditions and to uncertainties related to some model parameters. Considerable gains in time, money and operator exposure can be achieved using this predictive code.

7.1.3.2 LLWAA code description

Further description refers to the Belgian situation where the plant process waste (ion-exchange resins, evaporators concentrates, and spent filters) is conditioned at the plant while the technological waste is only pre-treated. This waste is transported on the Belgoprocess site for further processing in the CILVA facility, the centralised facility for treatment and conditioning of Belgian low level waste. The conditioned waste is stored in dedicated buildings located on the same site.

LLWAA is a full featured user friendly computer program, enabling the waste producer to determine the critical nuclides inventories in nuclear power plant radwaste streams and to declare the isotopic content of each waste package.

The LLWAA code calculates the radionuclide inventories in the process and technological nuclear power plant (NPP) waste on the basis of characteristics and operating conditions of the NPP.

LLWAA output deals with:
- The activities of the critical nuclides in the Reactor Coolant System and in the Nuclear Auxiliary Systems;
- The activity inventories of those nuclides in the on-site conditioned process waste streams and in the non-conditioned technological waste streams;
- The scaling factors in the on-site conditioned and non-conditioned waste streams.

The LLWAA code is site-specific as it takes into account the design characteristics and operating conditions of the different units of the site with a centralized waste treatment facility. The calculation of the activity inventories for each waste package requires, for each unit, the proper modelling of the Reactor Coolant System (RCS), the Nuclear Auxiliary Systems (NAS) and the Liquid Waste Processing System (LWPS).

Basically, the code parameters are classified into three categories:
- Category 1: parameters related to the operating conditions which can vary from one cycle to another. The values of these parameters are the main input data of the code and are based on the routine reactor coolant measurement results (Co-60, Cs-134, Cs-137, I-131, I-133, I-134 activities and Cl- concentration). The type of fuel (MOX or U-235 enriched fuel), fuel cycle length or rated plant power are additional input values;
- Category 2: parameters pertaining to the plant basic characteristics and operating procedures. Among those we find RCS, NAS and LWPS flow rates, boron content in the RCS and evaporators ...;
- Category 3: core activity inventories (calculated by means of the ORIGEN-2 code), chemistry of the primary coolant, equipment materials and surface areas, equipment corrosion rates, decontamination factors of the ion-exchange resins and filters, physical-chemical forms of the activity in the RCS (fraction of activity present in soluble and in insoluble forms) during normal operation and refuelling outage, diffusion coefficients through fuel cladding defects.

As long as the plant design characteristics and operating conditions remain unchanged, category 2 and 3 parameters are not to be modified.
The code output consists of:
- The specific activities (or activity inventories) of the critical nuclides in the RCS, ion-exchange resins, filters, evaporator concentrates and miscellaneous on-site non-conditioned waste (combustible waste, non combustible waste, compactable waste, ventilation filters, ...);
- The scaling factors of the critical nuclides in the same waste streams and waste packages.

Table 3
Typical activity inventory given by LLWAA for a NPP RCS

Isotopes	Activity [MBq/t]	Isotopes	Activity [MBq/t]	Isotopes	Activity [MBq/t]
C-14	2.06E-02	Sr-90	1.41E-03	U-238	5.00E-09
Cl-36	8.73E-05	Tc-99	1.84E-05	Pu-238	1.16E-04
Mn-54	1.36E+01	I-129	1.59E-07	Pu-239	8.99E-06
Fe-55	1.22E+01	Cs-134	7.18E-02	Pu-240	1.69E-05
Co-58	9.81E+01	Cs-135	2.32E-07	Pu-241	4.29E-03
Ni-59	4.26E-02	Cs-137	5.00E-02	Am-241	1.64E-05
Co-60	1.00E+01	U-234	1.91E-08	Pu-242	7.24E-08
Ni-63	3.31E+00	U-235	5.15E-10	Am-243	1.44E-06
Nb-94	1.37E-02	Np-237	4.13E-09	Cm-244	1.44E-04

The LLWAA code structure is on figure 7.3.

The predicted values were compared to the values derived from specific measurements performed by the Belgian Nuclear Research Centre at Mol (SCK□CEN). There is good agreement between predicted and measured values for most of the critical nuclides.
The code is qualified and accepted by ONDRAF/NIRAS, the Belgian Agency for Radioactive Materials, after an in-depth critical review performed by AVN, an organisation licensed by the Belgian Government in the field of Nuclear Safety.

7.1.3.3 LLWAA-DECOM code

LLWAA-DECOM code [7.17] is based on the LLWAA code calculating the required specific activities as input data.

The interrelationship of both codes is given in the figure 7.4.below.

The costs of the future decommissioning of the NPPs depend on the equipment contamination (pipe-work, valves, heat exchangers ...). Therefore a code has been developed to assess the critical nuclides activities deposited on the equipment of the nuclear auxiliary systems. This code takes into account the contamination in the streams of the systems (calculated by LLWAA), the operating conditions (fluid velocity, pH, and temperature), the corrosion products characteristics (particulate diameter distribution) and the nuclides deposition/release rates on the equipment.
The main goal of the program is to estimate, by using correlation factors, the deposited activity on the piping in contact with radioactive liquid or gaseous fluids. Scaling (correlation) factors address only isotopes having the same generation mechanism and behaviour in the reactor coolant system. For fission products the reference isotope is 137Cs, for activation products 60Co is the reference.

As input to the LLWAA-DECOM code, a number of parameters are considered:

- Characteristics of the equipment to be dismantled (piping diameter, pipe rugosity, ...);
- Operating conditions (temperature, average fluid velocity, pH, number of cycles, cycle life, ...);
- The corrosion products characteristics (particle density, particle diameter distribution, ...);
- Physical and chemical characteristics of the isotopes (decay rate λi, decay energy photon conversion factors hE, ...);
- Specific activities of the considered isotopes Cv (calculated by the LLWAA code for the considered nuclear auxiliary circuit);
- Time elapsed between the reactor final shutdown and the decontamination or dismantling.

Table 4
Example of the input table of the LLWAA-DECOM Code
Hot part of the steam generator tubes

NPP'S information	
Number of operation cycles	20
Cycle length (months)	10
t (t_decom. - t_shutdown)(years)	40
t (t_decont. - t_shutdown)(years)	2

Particle properties	
Particle density (kg/m³)	2000

Pipwork properties	
Pipe diameter (m)	0,0132
Wall tickness (m)	0,0014
Pipe roughness (mm)	0,0015
Density material (Kg/m³)	7860
Distance Detector - pipe (mm)	22

Thermal and hydraulic conditions	
Absolute temperature (°K)	574,15
Average fluid veloctiy (m/s)	2,57

Cl-36	2,30E+02
Mn-54	1,70E+06
Fe-55	5,40E+06
Co-58	8,80E+05
Ni-59	2,40E+03
Co-60	5,50E+05
Ni-63	2,90E+05
Nb-94	6,10E+00
Sr-90	8,90E+05
Tc-99	1,70E+04
I-129	1,40E+02
Cs-134	2,00E+07
Cs-135	1,50E+02
Cs-137	4,50E+07
U-234	3,30E+01
U-235	1,10E+00
Np-237	1,30E+00
U-238	6,60E+00
Pu-238	3,20E+03
Pu-239	4,70E+03
Pu-240	2,90E+03
Pu-241	5,00E+05
Am-241	6,10E+02
Pu-242	1,90E+00
Am-243	7,90E+01
Cm-244	9,00E+02
Fe-59	2,80E+05

Based on the input data, the code, using different modelling equations, calculates:

- The Brownian diffusion velocity (Vb);
- The particle velocity due to the fluid movement (Vf);
- The nuclides deposition and release rates.
- The latter are depending of V_b, V_f, of probabilistic settling of particles and of transport phenomena. Factors affecting the deposition and release rates are mainly the particle diameter distribution, the temperature of the fluid, the flow pattern (turbulent, laminar,) and the geometry (diameter, surface-to-volume ratio).

The evolution of the deposited activity depends mainly on:
- The nuclides deposition and release rates;
- The cycle life time and the number of cycles;
- The initial activity of the deposited materials at the start of the cycle;
- The specific activity of the considered isotope in the circulating fluid which is given by the LLWAA code.

The code main output consists of:
- The particle deposition and release rates;
- The deposited activities and the scaling factors at any given time.

Table 5

Example of output data of the LLWAA-DECOM Code

Deposited activities and scaling factors on the hot part of the steam generator tubes at reactor final shutdown and 40 years after reactor final shutdown

Deposited activities (Bq/m²)			Correlation factors		
Isotope	t = 0	t = 40y		t = 0	t = 40y
C-14	2,34E+06	2,33E+06	C-14/Co-60	2,82E-02	5,36E+00
Cl-36	4,90E+00	4,90E+00	Cl-36/Co-60	5,90E-08	1,13E-05
Mn-54	1,03E+07	8,43E-08	Mn-54/Co-60	1,24E-01	1,94E-13
Fe-55	6,37E+08	2,14E+04	Fe-55/Co-60	7,67E+00	4,92E-02
Co-58	2,44E+07	2,75E-55	Co-58/Co-60	2,94E-01	6,32E-61
Ni-59	5,11E+05	5,11E+05	Ni-59/Co-60	6,16E-03	1,17E+00
Co-60	8,30E+07	4,35E+05			
Ni-63	6,04E+07	4,58E+07	Ni-63/Co-60	7,28E-01	1,05E+02
Nb-94	1,30E+03	1,30E+03	Nb-94/Co-60	1,57E-05	2,99E-03
Sr-90	1,76E+04	6,69E+03	Sr-90/Cs-137	1,97E-02	1,88E-02
Tc-99	3,63E+02	3,62E+02	Tc-99/Cs-137	4,06E-04	1,02E-03
I-129	2,99E+00	2,99E+00	I-129/Cs-137	3,34E-06	8,40E-06
Cs-134	2,08E+05	2,83E-01	Cs-134/Cs-137	2,32E-01	7,95E-07
Cs-135	3,20E+00	3,20E+00	Cs-135/Cs-137	3,58E-06	8,99E-06
Cs-137	8,95E+05	3,56E+05			
U-234	2,11E+04	2,11E+04	U-234/Cs-137	2,36E-02	5,93E-02
U-235	7,03E+02	7,03E+02	U-235/Cs-137	7,85E-04	1,97E-03
Np-237	8,31E+02	8,31E+02	Np-237/Cs-137	9,28E-04	2,33E-03
U-238	4,22E+03	4,22E+03	U-238/Cs-137	4,72E-03	1,19E-02
Pu-238	2,00E+06	1,45E+06	Pu-238/Cs-137	2,23E+00	4,07E+00
Pu-239	3,00E+06	3,00E+06	Pu-239/Cs-137	3,35E+00	8,43E+00
Pu-240	1,85E+06	1,84E+06	Pu-240/Cs-137	2,07E+00	5,17E+00
Pu-241	2,78E+08	4,02E+07	Pu-241/Cs-137	3,11E+02	1,13E+02
Am-241	3,88E+05	7,92E+06	Am-241/Cs-137	4,34E-01	2,22E+01
Pu-242	1,21E+03	1,21E+03	Pu-242/Cs-137	1,35E-03	3,40E-03
Am-243	5,05E+04	5,03E+04	Am-243/Cs-137	5,64E-02	1,41E-01
Cm-244	5,13E+05	1,10E+05	Cm-244/Cs-137	5,73E-01	3,09E-01
Fe-59	3,38E+06	7,10E-93	Fe-59/Co-60	4,07E-02	1,63E-98

Direct measurements of deposited activities present a number of difficulties. Therefore, to validate the code, a dose rate model, based on the principle of consistency between calculated and measured dose rates values, has been coupled to the calculated deposited activities.

The predicted values of the scaling factors are compared to the derived values from specific dose rate measurements performed during NPP shutdown and decontamination operations.

There is a general good agreement between predicted and measured contact dose rates. The LLWAA-DECOM code allows the calculation of the scaling factors within the range excellent to good. There is also a good consistency between the measured and predicted

values of the deposited activities on the portions of the systems when they were measured, namely during the steam generator replacement programmes.
Some precautions have to be taken in order to obtain valid measurements:
- The measuring point should be representative for the equipment to be considered;
- 'Dead' zones or drain tie-ins should be avoided for measurements (risk of 'hot spots');
- Back-scattering effects of other equipment in the neighborhood of the equipment to be measured should also be avoided;
- Accurate knowledge of the distance between equipment and detector is required.

7.2 *Measurement methods for radiological characterization of waste*

The large volume of materials generated from decommissioning of nuclear facilities is a big concern. The main evacuation routes for dismantled materials are 'low level' nuclear waste, disposal, recycling and reuse of materials. Because concrete and steel are valuable materials and because of the need to reduce waste directed to radioactive disposal facilities, the 'recycling and reuse' evacuation route has been carefully studied in the past 20 years. The recycling options for rubble concrete are typically to use it in civil engineering for road construction or as an additive for new concrete. Metal can be sent in melting facilities. The consequence of the melting process is, firstly a decontamination of the metal as caesium 137 is volatilised from the metal and secondly a more homogeneous distribution of the remaining radioisotopes, which permits more accurate measurements of radioactivity.

Material can be released from regulatory control for the recycling (metals and building rubble) or reused (metal tools, building), if the activity per unit mass contained in the material is lower than the clearance level. In order to verify compliance to the very low clearance level, the detection limit of the measurement monitor has to be very low. This means that the detectors have to be well-shielded and that the measurement time has to be rather long.

This chapter will focus on clearance measurement of Exempted Waste (EW) rather than Low or Intermediate Level Waste (LILW) and High Level Wastes (HLW). Some of the monitors, like gross gamma counting monitor or gamma spectrometry monitor, used for the radiological characterisation of EW can also be used for LILW. As the clearance level are much lower than the levels for LILW, the measurement time to detect the EW level is longer and the measurement methods have to be more precise.

7.2.1 Clearance of material from regulatory control

7.2.1.1 International recommendations

In 1984, IAEA in co-operation with the OECD/NEA, started a programme with the specific objectives of developing principles for exemption of radiation sources and practices from regulatory control. As part of this work exclusion, exemption and clearance were defined **see figure 7.5.** A radioactive source can either be excluded from regulatory control, because of its nature or exempted from regulatory control because of the low level of activity involved in the practices. When materials are under the regulatory control, they can only be released if the level of activity per unit mass is below the Clearance Level.

The aim of the international recommendations is to look at the health and the environmental impact of the clearance option in order to minimise the radiological risks to the workers and the public. In 1988, the Safety Series N°89 [7.18] that was issued jointly by the IAEA and the OECD-NEA suggests:
- a maximum individual dose/practice of **10 μ Sv/year**,
- a maximum collective dose/practice of 1 manSv/year,

to determine whether the material can be cleared from regulatory control or if other options should be examined. Those values applies to practices (nuclear sector) and not to work activities (Naturally Occurring Radioactive Material - NORM industry).

Methods to derive those values into Clearance Level (activity concentration - Bq/g) for specific isotopes using scenarios and pathways are presented, for instance in the Safety Series No. 111 P1-1 [7.19] and in the Radiation Protection 122 part I [7.20] and part II [7.20b].

Not only the nuclear risks but also the 'classic' risks have to be evaluated when comparing the impact caused by mining and milling processes due to the replacement of steel and concrete and the clearance option.

The main guidelines and recommendations are:
- Safety Series No. 89 [7.18] that lays down the Principles for the exemption of radiation sources from regulatory control.
- IAEA TEC DOC 855 [7.21] that recommends a set of unconditional Clearance Levels (in solid material).
- IAEA TEC DOC 1000 [7.22] that presents clearance of materials resulting from the use of radionuclides in medicine, industry and research.
- Radiation Protection 89 [7.23] that recommends radiological protection criteria for the recycling of metals from the dismantling of nuclear installations.
- **The European council directive 96/29** [7.24] which member's states agreed to implement into their national legislation by the year 2000. However, this Directive does not prescribe the application of Clearance Levels by competent authorities. So until now, no international harmonisation of those Clearance Levels has been reached. This Directive introduces the concept of work activities, involving exposure to natural radiation sources, typically applying to NORM industry. The European Recommendations RP-122 has been established by the group of experts established under article 31 of the Euratom Treaty:
  - Radiation Protection 122 (part I) [7.20] that gives practical use of the concepts of clearance and exemption for practices (recommendations of the Group of Experts established under the terms of Article 31 of the Euratom treaty).
  - Radiation Protection 122(part II) [7.20b] that gives practical use of the concepts of clearance and exemption for work activities.
    - As a result of the large volumes of material processed and released by NORM industries, in reality, the concept of exemption and clearance merge, and so it is appropriate to lay down **a single set** of levels for both exemption and clearance;
    - While the basic concept and criteria for exemption-clearance for work activities are very similar to those for practices, it is not meaningfull to define the levels on the basis of the individual dose criterion for practices (10 μSv per year). Instead, a dose increment, in addition to background exposure from natural radiation sources of the order **of 300 μSv** is appropriate.

The following table gives the Clearance Levels for Co-60 and Cs-137, recommended by some national and international documents. Those radionuclides are the ones commonly found in waste originating from decommissioning of PWR reactors.

Table 1
Clearance Levels for Co-60 and Cs-137

Radio element	Belgium legislation	RP 122	RP 89 metal scrap	TEC Doc -855 derived [illegible]	SSK[1]
Co-60	0.1 Bq/g	0.1 Bq/g	1 Bq/g	Range from 0.1 to < 1Bq/g	0.1 Bq/g
Cs-137	1 Bq/g	1 Bq/g	1 Bq/g		0.5 Bq/g

7.2.2 Specific national approaches

As the application of clearance levels by competent authorities is not required, according to the Directive some countries like France chose not to release material from regulatory control. Some examples are given hereunder :

7.2.2.1 Belgium

The European council directive 96/29 was implemented in the Belgium national legislation (Royal Decree – 20th of July, 2001). The workers and public exposure from both nuclear and non-nuclear industries are regulated under the same legislation. The legislation gives a set of Clearance Levels for the recycling and reuse of solid nuclear waste. The Clearance Levels are similar to those recommended in the RP 122 (Part I, practices), except for Te-131m, Te-134, Re-186 and Np-239. The clearance levels for those radioelement are one order of magnitude lower than the levels given in RP-122.

7.2.2.2 Germany

The clearance levels are set forth in the new German Radiation Protection Ordinance (August 2001). A distinction is made between unconditional clearance and cases in which the issuing of clearance is directly linked to the use of the materials.

Several clearance type categories exist for conditional clearance and have specific limits. They comprise:
- solid materials and liquids for disposal at an incineration plant
- buildings to be demolished
- scrap metal for recycling.

Also, for unconditional clearance, different sets of limits are given for solid materials and liquids different from the following items:
- building rubble and excavated earth (more than 1000 ton/year)
- buildings for continued and renewed use.

[1] Clearance of materials, buildings and sites with negligible radioactivity from practices subject to reporting or authorisation' recommended by the "Strahlenschutzkommission" (SSK).-

7.2.2.3 Switzerland

For conditional clearance, the release of solid waste materials with specific activities not exceeding 100 times the exemption limit (Annex 3 to the Radiation Protection Ordinance) may exceptionally and pending agreement of the Authority be released into the environment. However, this cases require guaranties that by mixing them with inactive materials, the limits set forth in the Annex 2 to the Radiation Protection Ordinance will not be exceeded.

7.2.3 Clearance procedure

Optimization the development of clearance methodologies in a decommissioning project is extremely difficult to achieve. However, respecting the following steps should help to establish methodologies that individually ensure compliance to the clearance level:

- preliminary survey
- development of methodologies
- selection of the instrumentation
- validation of the instrumentation
- quality insurance
- material management

7.2.3.1 Preliminary survey

In a decommissioning project of a nuclear facility the preliminary planning is a key step. During this planning, the inventory and distribution of the radionuclides likely to be present in various parts of the plant and sites are studied. These data are obtained through:

1. a good knowledge of the plant and its process streams,
2. theoretical calculations of induced activity
3. measurement samples taken during operational and maintenance tasks.

This information is usually supplemented after the final shutdown of the plant by a preliminary monitoring survey of the facility to confirm previous records, supply additional data and look for hot spots or non-uniform distribution of activity.
This preliminary information will also be used to decide which plant components are likely to be suitable candidates for recycling or reuse. For example, if preliminary core samples or calculations show that the concentration of activation products in steel is above an acceptable level the material would no longer be considered for recycling.

7.2.3.2 Development of methodologies

The preliminary study describes, among others, the inventory and the characterisation of the material candidate for clearance. Base on that study, clearance measurement methodologies are developed for each category of material. A precise characterisation of the material to be measured is a key issue to define a suitable measurement methodology and to select the appropriate instruments. Wrong assumptions of the isotopic ratio and of the absorption factor are the main sources of measurement error.

7.2.3.2.1 Categorisation of material

Materials are classified by selecting a group of materials according to their origin, the physical and chemical nature and the isotopic ratio.

The categories are made keeping in mind the abilities of the instruments available on the market. For instance for a specific group of materials, the isotopic ratio can be difficult to assess but it can be proven that no pure alpha or beta emitters are present. In that case, an instrument based on gamma spectrometry is probably the most appropriate instrument and a pre-analyse to define exactly the isotopic ratio is no longer needed.

The group of material should be characterised, describing:

- the scope: the identification of the material and the amount of material.
- the history of the material, possible incidents
- the decontamination process applied and its consequences
- the characterisation of the material; its nature, physical conditions (solid, porous, powdery, fibrous), compositions and shape of the components.
- the radiological characterisation;
    - the radioisotope spectrum most often established by sampling and analysed in situ or in laboratory.
    - the nature of the radioactivity and its fixation on the material (fixed and/or transferable surface contamination, activity gradient along the thickness of the part, chemical form, activation beneath the surface) and the heterogeneity of the radioactivity distribution.
- the non-radiological risk.
- the Clearance Level; reference to the set of Clearance Levels in the legislation or call for an authorisation to the competent authority (documented by an impact study).

Measurement methodologies are developed for each category. The description defines the scope of application of the methodology and gives the assumptions for the validation.

According to the Belgium legislation, the measurement methodologies have to be approved by the health physics officer and by the competent national authority.

Development of methodologies

A measurement methodology is a measurement procedure applied to verify compliance with the clearance levels. Based on the descriptions given in the characterisation of the material the most appropriate instruments are selected. Direct measurements are done in-situ or in a measurement area dedicated to clearance measurement. That area is often selected for its low level of background. Most of the samples taken are measured in laboratories.

The method should focus on measuring directly the most abundant radioelements present. Gamma and beta emitters of relatively high energy are quite easy to measure directly whereas alpha emitters and low gamma and beta energy emitters can be more difficult to measure directly with a good certainty. The activity of radionuclides, which are difficult to measure directly, can be assed using the isotopic ratio. For example, ^{55}Fe and ^{63}Ni can often be correlated to ^{60}Co and ^{90}Sr to ^{137}Cs. In that case, the ratio must be checked more frequently by laboratory analysis or spectrometric survey instruments.

Because it is unlikely that the contamination and/or the activation will be spread homogeneously, it is essential to define the appropriate mass or material surface over which radionuclide quantities may be averaged.

The characterisation of the material is used to define corrections factors due to the absorption by the material to be measured, absorption by a layer of dust on material (typical for direct contamination measurement), retrodiffusion effect, etc. For components in which it is likely that the contamination could have penetrated into the material, for example soil or concrete, core samples or borehole monitoring would be required to confirm compliance both at depth and on the surface.

The distribution of measurement and sample points must have a statistically sound basis such that the results demonstrate compliance with an adequate level of confidence. Sampling points whether they are for in situ measurements, swabs or samples, can be selected on the following statistical basis:
- random sampling,
- stratified random sampling (i.e. random within a stratum or survey unit)
- systematic sampling (based on the grid system).

The extra actions to take due to specific conventional risks such as asbestos should be taken into account in the methodology.

7.2.3.3 Selection of the instrumentation

Most of the instruments available on the market to carry out clearance measurement or measurements for preliminary study are described in chapter 7.2.4.They should be selected according to there detection limit in the background level expected at the place of the measurement. Technical aspect of the instrument like security systems for door openings, weight, volume of the measurement room, decontamination possibility of the measurement room, etc should be carefully studied with the local relevant technical experts.

7.2.3.4 Validation of the methodology

The principle of the validation is to prove that in the conditions described in the characterisation, the detection limit of the instrument is lower than the clearance level. The validation evaluates the error of the measurement due to statistical errors of measurement (see chapter 2.5), systematic error mostly due to errors of assumption and errors due to the sampling procedure (representatively of the sample). The validation should be based on standard and norms.

Precise definition of the measurement conditions is a way to set the correct assumption and therefore decreasing the errors. An accurate characterisation does not only have consequences on the errors of measurement but also on the constraint ('heaviness') of the methodology. Indeed, as a principle of conservatism, worst case assumptions have to be considered when the conditions are unknown. For instance, when measuring with an handheld monitor, an unknown ratio of Co-60/cs-137, the worse case assumption is to consider a contamination of 100 % of Co-60, Co-60 having a lower detection efficiency than Cs-37. The consequence is that the alarm level will be lower and the measuring time will be longer.

7.2.3.5 Quality management

The purpose of a quality assurance (QA) on monitoring for compliance with Clearance Level is to ensure that sampling, analysis, monitoring, documentation, interpretation and use of data generated for this purpose will not result in the release of a component that could constitute a public health hazard. The importance of having a well thought-out QA cannot be overemphasized. The QA must start with the detailed action plan and be an essential part of every step until the component has been released for recycling or reuse. The reader is referred to documents such as ref. [7.32] for further information.

7.2.3.6 Material management

A management programme has to be developed for the accounting of material cleared. See chapter 6 en [7.30].

7.2.4 Instrumentation – measurement device

Clearance measurements can be carried out by direct measurement on the material (non-destructive method) and indirect measurement (sampling or smear test). The main type of instruments used to carry out non destructive measurements are:

- Hand held monitor;
- Gross gamma counting with plastic scintillators;
- High resolution gamma spectrometry with HPGe
- Low resolution gamma spectrometry with NaI
- Gamma cameras (spectrometry or imaging).
- Neutronic analysis.

The Clearance Levels are just a small fraction of the natural background level, so it is essential that the detection limit of the instrument is lower than the Clearance Level. Only instruments with low detection limit will be considered for measurement at Clearance Level. A common practice is to evaluate, in the given condition the needed duration of measurement to ensure that the detection limit is lower than the Clearance Level.

7.2.4.1 Hand Held monitor

The probe of the hand held monitor is mostly a gas filled or scintillations detectors. The measurement methods is simply to place the probe in front of the material to control. The duration of the measurement is long enough so that the detection limit is lower than the clearance level (The concept of detection limit will be further developed in chapter 7.2.5.

7.2.4.1.1 Scintillation detector

In scintillation detector, the detection of the radiation occurs in a scintillation detector. When the radiation passes through the scintillation detector, emission of very low intensity light (only few photon) is generated. This very low intensity of light is amplified and transformed in electronic impulse by a photomultiplier tube. The electronic signal is then treated to count the number of event per second. A calibration factor is used to evaluate the activity or activity per surface unit (taking into account the surface of the probe).

To detect alpha radiations, a ZnS(Ag) scintillation powder product is generally used whereas plastic is used to detect beta radiations. More precisely, the commonly called "plastic" used to detect beta particle is made of a crystal of Anthracene type ($C_{14}H_{10}$), Stilbene ($C_{14}H_{12}$) or plastic (NE 102 or NE 105). Because the maximum range of an alpha particle is very low, the alpha detector is a very thin layer see **figure 7.6.**

Those scintillators can be used separately or simultaneously. The probe is then respectively called a 'pure probe' or a 'dual probe'. In dual probe, a thin layer of ZnS(Ag) to detect alpha is placed on top of the plastic scintillator used to measure beta particles. Because the thin ZnS(Ag) layer absorbs a part or all the beta emitters of low energy, it is preferable to use a pure beta probe to control material contaminated with beta emitters of low energy such as Co-60. Radioelements like C-14 can not be detected as beta radiation are completely absorbed by the ZnS(Ag) scintillation detector.

As a consequence, a simultaneous measurement of alpha and beta contamination:

- would be a long measurement (10 to 50 seconds) when the material is mostly contaminated with beta emitters of low energy such as Co-60.
- would not be possible if the beta emitters to be detected have an energy lower than 200 keV.

The response due to an alpha source in the beta channel can be quite important (interference). However state of the art hand-held detector applies a correction factor to reduce the interference.

7.2.4.1.2 Gas filled detector

A gas-filled detector is basically a chamber containing a gas and two electrodes. The absorption of radiation in the gas results in the production of ion pairs. A voltage applied between two electrodes causes the negative ions to be attracted to the positive electrode, and positive ions to the negative electrode. The flow of ions to the respective electrode produces an electric current, which is measured. According to the voltage applied to the electrodes, the chamber works in various modes. The most commonly used for clearance measurement is the proportional mode **see figure 7.7.**

7.2.4.1.3 Calibration and good practice for hand held monitor

After setting the optimum High Voltage to the photomultiplier, the hand held monitor is calibrated with Class 2 reference sources of C-14, Co-60, Cs-137, Cl-36, Sr-90 + Y-90 and Am-241. The Class 2 reference sources are made according to the ISO 8769 [7.25], and comply with the requirements for traceability to NIST specified in the American National Standard ANSI N42.22-1995 [7.26].

The instrument efficiency is evaluated according to ISO 7503-1 [7.27]. The instrument efficiency is the ratio of the net indication on the instrument and the surface emission rate at the surface of the source, in given geometrical conditions. The 4π efficiency is defined as half of the efficiency of the instrument. The following table gives an overview of the 4π efficiency of a proportional counter, an alfa/beta probe and pure beta probe:

Table 2
Energy – 4π efficiency Label on Dual and Beta probe

	Gaz- PC	Pl. α/β	Pl. β
Carbon-14 ($E_{\beta max}$ = 0,154 MeV)	20 %	0 %	15 %
Cobalt-60 ($E_{\beta max}$ = 0,31 MeV)	30 %	10 %	32 %
Caesium-137 ($E_{\beta max}$ = 0,51 MeV)	34 %	25 %	42 %
Chlorine-36 ($E_{\beta max}$ = 0,71 MeV)	36 %	34 %	48 %
Sr-90/Y-90 ($E_{\beta max}$ = 2,26 MeV)	36 %	37 %	49 %
Americium-241	24 %	27 %	0 %

A control source is allocated to each probe. The control source is an alpha and/or beta emitter. After calibration, this source is measured to establish the expected value in counts per seconds. The ISO 7503-1 [7.27] recommends that the instrument should be controlled daily during measurement campaigns. It also assesses that a deviation higher than 25 % should lead to the re-calibration of the instrument. (We use criteria of 20 % for alpha emitters and 10 % for beta emitters.)

The procedure to carry out clearance measurements is inspired by the ISO 11932 [7.28] and ISO 7503-1 [7.27].

7.2.4.2 Gross gamma counting (Dose rate monitoring and related activity evaluation)

Gross gamma counting monitor often includes large plastic scintillation material with embedded PMT mounted inside a shielded box **see picture 7.1**. A geometry of 6 plastics is needed for detection in a 4π solid angle. When using only 2 detectors, the activity measured depends strongly on the location of the contamination. For the same level of activity, the signal will be low if the contamination is just in between the 2 detectors and high if it is located very close to one of the detectors.

Plastic scintillation detectors are very sensitive to gamma radiations. This results in a high efficiency but also a bad ratio signal to background. Therefore, to thwart this disadvantage, the detectors are shielded from external source or located in a low background environment.

The gamma radiations emitted from the waste are detected by the plastic scintillation material. The pulse of light produced by each event in the scintillation detector is measured by a photomultiplier. The total count rate of pulses above a certain energy threshold produced is used to determine the total activity due to all gamma-emitting radionuclides present in the waste to be measured. No information is given on the energy of the gamma radiations and therefore on the radioelements present. This phenomenon justifies the name: "Gross gamma counting monitor". However, it is possible to improve the system and upgrade it to detect Cobalt-60. This happens by fixing two thresholds and therefore creating a large Region Of Interest (ROI) around the energy of the two gammas emitted by Cobalt-60. The activity of ^{60}Co can also be evaluated by the Cobalt Coincidence Method. These methods conclude on the detection of ^{60}Co when two separated scintillation detectors simultaneously detect a γ.

The detection efficiency is quite constant when the energy of the emitted gamma is higher than some 100 keV. But the efficiency for a given radioelement strongly depends on the number of gamma emitted per disintegration. For instance, the efficiency for Cobalt-60 that emits two gamma per disintegration is a factor two higher than Caesium-137 that emits one gamma per disintegration. The global efficiency, used to calculate the activity present in the waste, is evaluated for an assumed ratio (defined mixture) of radionuclides. The measured value is only trustable if the ratio is reasonably constant for all the waste being measured. As a principle of conservatism, the worst ratio of radioelement is assumed. Of course, this leads to an overestimation of the activity present in the waste.

When the instrument is equipped with a weighing platform the activity concentration (Bq/g) can be evaluated.

To summarize, the use correctly a 'gross gamma counting' monitors, the following should be considered:

1) Precise knowledge of the ratio of radionuclides.
2) A 4 π geometry detection (6 detectors) is less sensitive to the location of the contamination.

7.2.4.3 High resolution spectrometry with High Purity Germanium (HPGe) [7.31]

If high purity germanium detectors are used instead the plastic scintillation detector, the impulsion created in the detector is more closely proportional to the energy of the gamma emitted by the radioelement present **see picture 7.2**. It is therefore possible to identify the radioelement present in the drummed or boxed waste placed inside a shielded cabinet. Because of the small band gap of germanium (0.7 eV), it is conventional to operate such detectors at liquid nitrogen temperature (77 K).

As opposed to plastic scintillator, large volume HP-Ge can not be manufactured. HP-Ge detector will typically have a volume between 10 to 30 cm3 for measurement of low gamma energy or in cylindrical shape for high gamma energy. Therefore a 4π geometry measurement can not be carried out with such a detector. This means that the response of the detector can depend on the location and the distribution of the activity in the container and on the absorption by the material. To have a more representative measurement and avoiding evaluation errors of the activity level, the measurement should be carried out in several locations over the containers.

To achieve this, the drum is placed on a rotating table and either a few fixed detectors measure the drum, or a moving detector scans the fixed container.

The monitors are most of the time equipped with a weighing system to evaluate the activity per unit mass. A shielding zone will be used to reduce the background level.

To calibrate the system, nuclide specific activity can be calculated using reference drums filled with different inactive homogeneous materials with different densities. Linear radioactive sources can be introduced in the calibration drum, each in the centre of concentric shells of equal volume. This positioning approximate a homogeneous distribution of the activity.

As a conclusion, when using a 'gamma spectrometry' monitor:
1) This monitor gives the activity per unit mass for each radioelement which can be directly compared to the clearance levels.
2) Due to the measurement geometry, the main source of error is the activity distribution. This error increases when measuring low level gamma energy.

7.2.4.4 Low resolution gamma spectrometry
Mostly sodium iodide NaI (Tl) will be used to carry out LRGS. NaI detectors have a much poorer energy resolution than HPGe detectors, but are suitable for use when the gamma-ray spectra are relatively simple. They can be manufactured in larger volume than HPGe and they are cheaper and require less maintenance than an HPGe detector.

LRGS systems also tend to use multiple or scanned detectors to measure a rotating waste. Very high efficiency, low background LRGS assay systems can be produced either by carefully shielding a few detectors close to the measured item or by building NaI detectors into a low background shielded enclosure.

7.2.4.5 Camera's spectrometry or imaging
A system of camera's or a transportable monitor have been developed. They are mainly used in the preliminary phase for the characterisation and the grouping of material per category.

This can be:
- A HPGe monitor cooled by nitrogen. The detector is collimated to measure the radiation emitted by the waste in a specific geometrical angle. The activity level is estimated by defining the composition of the waste (identification of the density of the material and the location of the activity) and then using a mathematical code. This monitor will give information on the activity level and the radioelement present.
- Plastic scintillation detector associated with an imaging system. The collimated detector scan the area to be measured. On each point of measurement, the activity level is measured and a map of the activity level can be created. This mapping can be associated to the picture of the area scanned for a better visualisation of the activity distribution in the area of interest.This kind of instrument is often used in the pre-study for identification of the 'hot spot'.

7.2.4.6 Neutronic analysis

7.2.4.6.1 Passive neutron measurement
Passive neutron measurement techniques are not generally applicable to clearance measurement problems because, for the relevant isotopes the clearance level is set by the alpha emission rate which is much higher than the neutron emission rate: Neutron emission arises from spontaneous fission, from (α,n) reactions following alpha decay or from delayed neutron emission by fission products following spontaneous fission. For example, consider a 200 liter drum containing 200 kg plutonium contaminated material (PCM). For reactor grade Pu, the 0.1 Bq/g clearance level would allow just 0.04 μg Pu in such a drum. Its neutron emission rate would be ~ 0.95 x 10^{-5} s^{-1} or one neutron every 7 hours. The best passive neutron detection limit for such drums is now ~ 0.5 mg, using sophisticated equipment installed.

For uranium waste streams the neutron emission rate per g is lower than for Pu, however its alpha emission rate per g is much higher. The 0.1 $Bq.g^{-1}$ limit in a 200 kg, 200 liter drum is now equivalent to 0.23g U for 5% enriched material. With a neutron emission rate of 0.04 s^{-1}, this is practically undetectable by passive neutron counting.

7.2.4.6.2 Active neutron interrogation

Active neutron devices comprise a neutron source (usually a DT tube producing 14 MeV neutrons), which is fired into a measurement chamber in short pulses (order of milliseconds) at the sample to be measured. The configuration of the measurement cell ensures that fired neutrons are thermalised by polyethylene before striking the sample. The thermal neutrons striking the sample will cause fissions in any fissile material present resulting in the production of fast neutrons.

The measurement chamber is enclosed by walls of He-3 thermal neutron detectors clad in a layer of polyethylene and an outer layer of cadmium. The fast neutrons produced from fissions of any fissile material will be able to penetrate the cadmium passing into the polyethylene layer where they will be thermalised and subsequently detected by the He-3 detectors. As the fired neutrons are pulsed quickly, the presence of any fissile material can be determined by scrutinising the relationship between time after firing neutrons and the number of detected neutrons.

Commercially available systems can be used to quickly measure crates of drums for U-235. Generally use of this technique is mostly applicable where a bulk measurement of a large mass or volume of material is required. The technique is able to measure Uranium down to clearance levels. As this is a total fissile measurement, without Pu/U discrimination, clearance would need confirmation that no Pu could be present.

Active neutron interrogation requires the contaminant to be quite well known, i.e the nature and the part of the fissile isotopes. Moreover, the material of the waste can greatly influence the results of the measurement, especially if neutron absorbers (Chloride for example) are likely to be present. This method requires also a neutron source (neutron generator or neutron emissive isotope), and many neutron counters. Hence it is not simple to operate and is rather expensive. To conclude, this technique is theoretically applicable, in certain cases, but has not been operated for clearance level waste control yet.

7.2.4.7 Sampling and laboratory equipment

Sampling techniques are often used in the pre-study phase in order to characterise the radioisotope spectrum or when other methods are impossible. The problem with the sampling technique is to extrapolate the result on to a bigger area. Using sampling measurement of a part of the material to be measured, as a substitute to 100 % direct measurements methods, is always difficult to validate from a statistical point of view. Taking that into account, samples will be taken where the risk of contamination is higher. For instance when sampling a wall, the sample would be taken where there are some cracks, as the risk of infiltration of the contamination is higher here.

The sample can be taken by diamond coring, by scrapping of the surface or by smear test.

The samples are then analysed in the laboratory either by alpha or gamma spectrometry to quantify the radioelement present (evaluation of the ratio) or by total beta or alpha counting to evaluate the level of contamination or activation in alpha and beta. These analysis are carried out directly on the specimen or after treatment (dissolution, separation, etc).

7.2.5 Uncertainty analysis – detection limit

The Clearance Levels are just a small fraction of the natural background level. Therefore to be able to verify compliance with the clearance level, it is necessary that the detection limit of the instrument is lower than the Clearance Level. Only instruments with low detection limit will be considered for measurement at Clearance Level. A common practice is to evaluate, in the given condition, the needed duration of measurement to ensure that the detection limit is lower than the Clearance Level. The detection limit is calculated according to the ISO 11929-1 [7.29].

For instance, for an hand held monitor:

Detection limit (cps) < Clearance Level (cps)

1. Detection limit (ISO11929-1) [7.29]:

$$\text{Detection limit} = \left(k_{1-\alpha} + k_{1-\beta}\right)\sqrt{R_0\left(\frac{1}{t_0}+\frac{1}{t_b}\right)+\frac{1}{4}\left(k_{1-\alpha}+k_{1-\beta}\right)^2\left(\frac{1}{t_0}+\frac{1}{t_b}\right)}$$

1. $k_{1-\alpha}$, $k_{1-\beta}$: function of probabilities α and β of committing an error of type I and type II, respectively; normal distribution,
2. R_0 : Back ground level (cps),
3. t_0 : duration of the BG measurement (s),
4. t_b: duration of the measurement (s).

2. Clearance Level (cps) = alarm level

$$CL(cps = CL(Bq/cm^2)xS_{vue}x\eta_{global}$$

1. CL : Clearance Level (Bq/cm2)
2. Svue: field of view by the detector (cm2),
3. ηglob: global efficiency of the instrument

The global efficiency of the instrument depend on the conditions of measurement. To define the alarm level and the measurement time, it is important to evaluate the measurement conditions, by characterising the material to be measured.

The characterisation is often described in a the preliminary study:

- the characterisation of the material; its nature, physical conditions (solid, porous, powdery, fibrous), compositions and shape of the components.
- the radiological characterisation;
  - the radioisotope spectrum most often established by sampling and analysed in situ or in laboratory.
  - the nature of the radioactivity and its fixation on the material (fixed and/or transferable surface contamination, contamination gradient in the thickness of the part, chemical form, activation beneath the surface) and the heterogeneity of the radioactivity distribution.

If the measurement conditions assumed are far from the actual conditions of measurement, the errors can be consequent.

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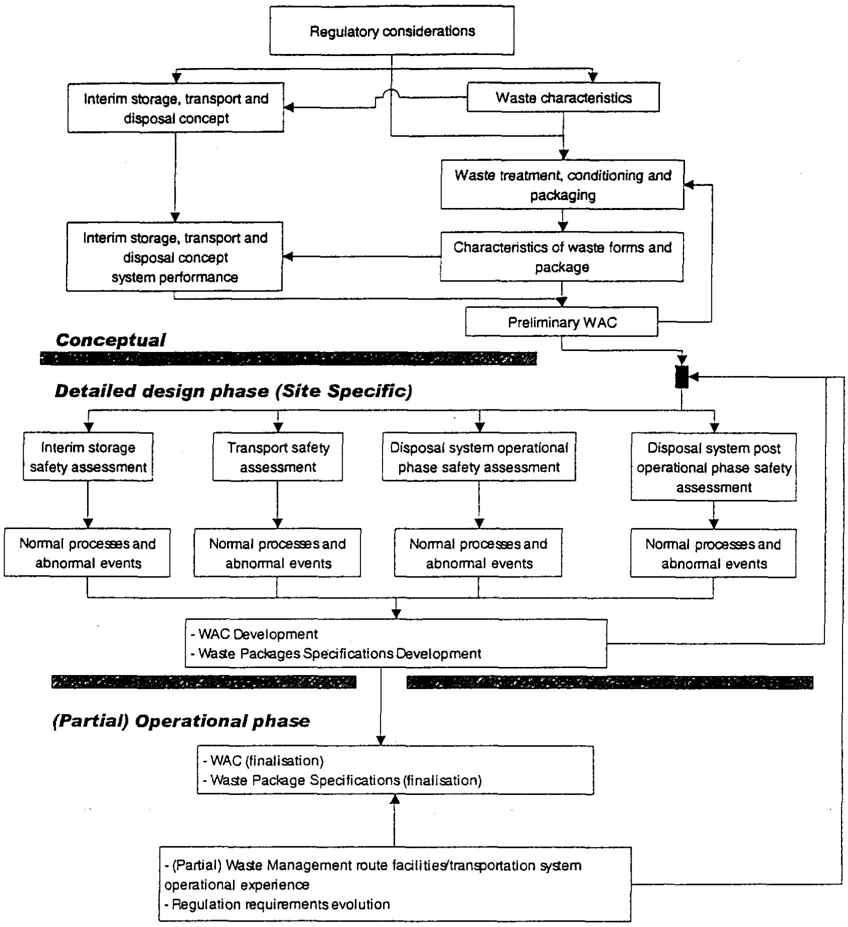
7.4. Figures and Pictures



Waste Acceptance Criteria e.g.

Radionuclide content	Structural stability/Voidage	Thermal effects	Swelling
Surface dose rate	Impact and fire resistance	Radiation effects	Organic materials content
Surface contamination	Leachability	Combustibility	Gas generation
Criticality Safety	Explosive materials	Pyrophoric materials	Microbial degredation
Hazardous waste content	Package identification	Package configuration (mass, dimensions, lifting and stacking features, etc.)	Free liquids
			Corrosion resistance
			Package venting

Figure 7.1 – General outline methodology for the development of Waste Acceptance criteria

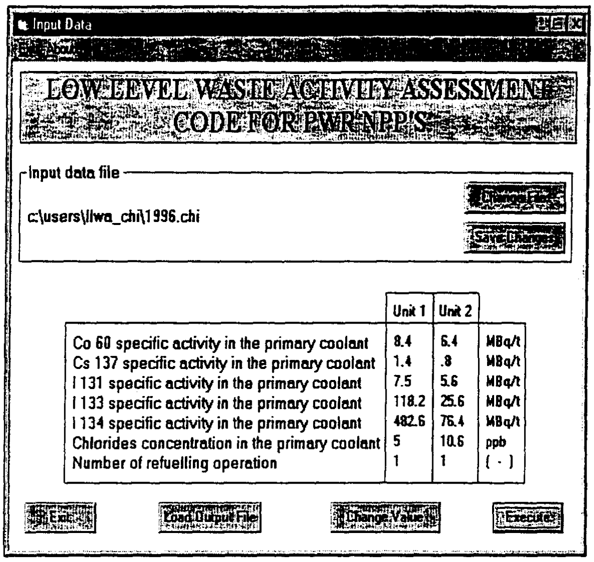


Figure 7.2.: Input window for the plant operating conditions

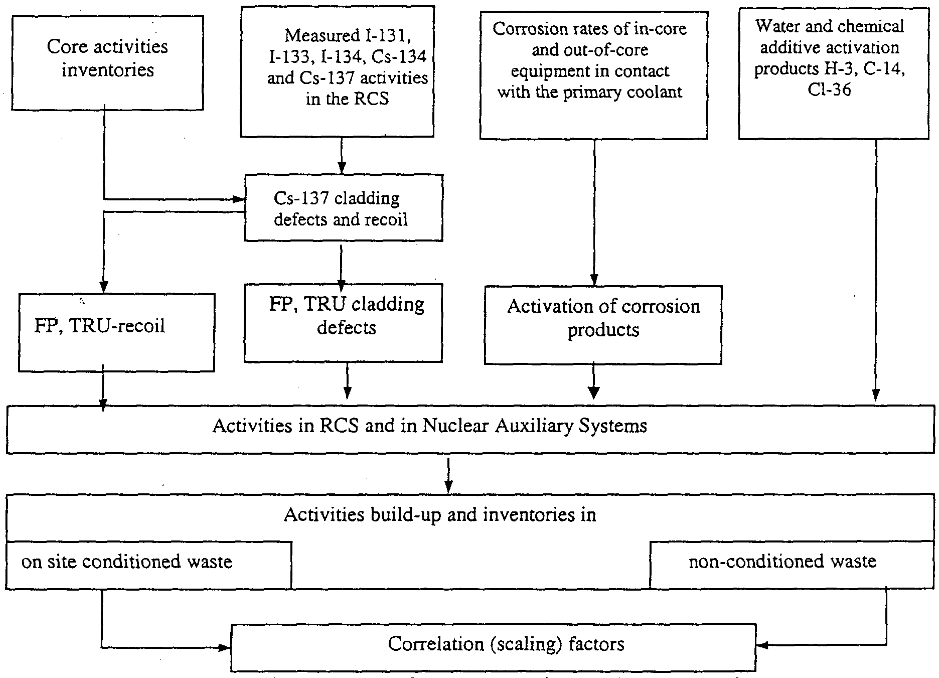


Figure 7.3.: Structure of the LLWAA code

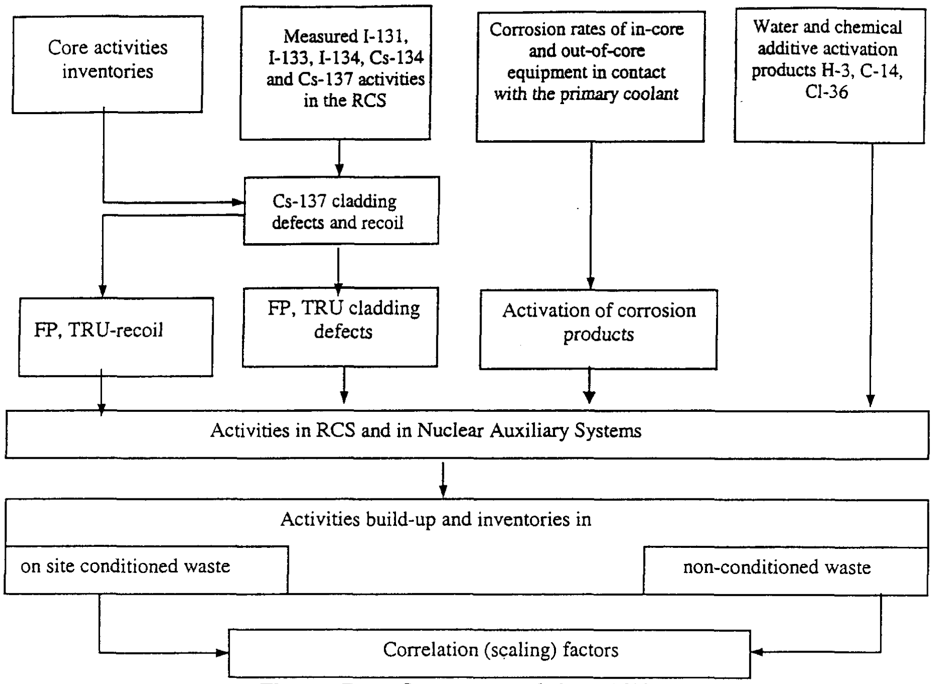


Figure 7.3.: Structure of the LLWAA code

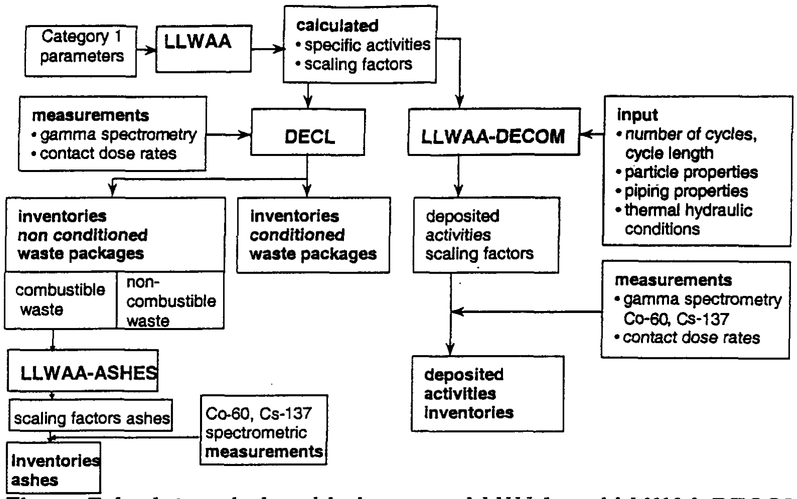


Figure 7.4. : Interrelationship between LLWAA and LLWAA-DECOM codes

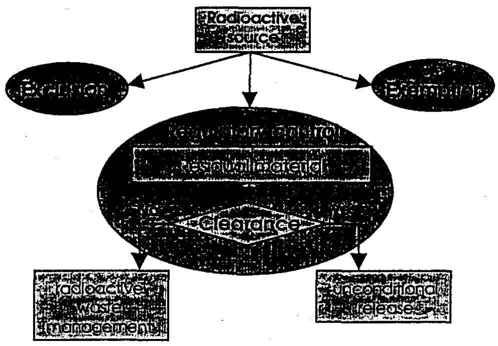


Fig. 7.5.: Concept of Exclusion, Exemption and Clearance.

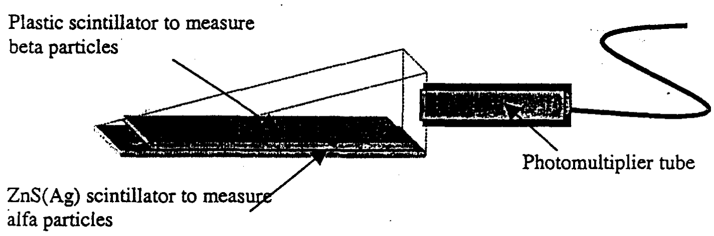


Fig. 7.6.: Dual probe hand held monitor

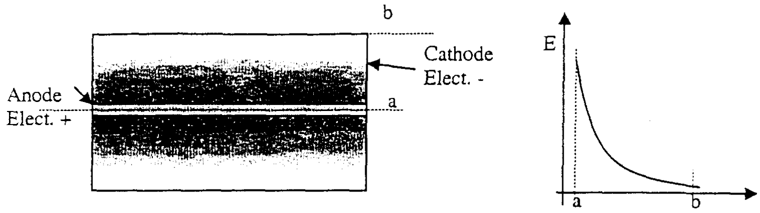
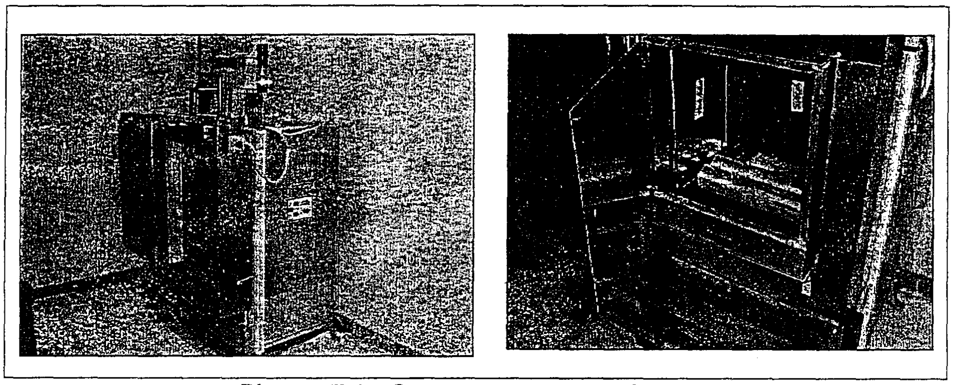
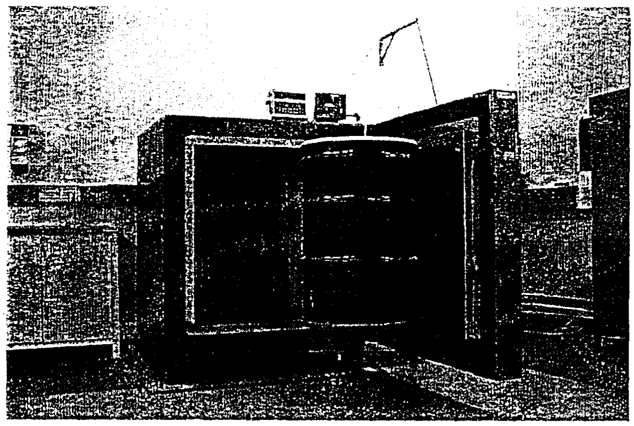


Fig. 7.7 Gas filled hand held monitor



Picture 7.1.: Cross gamma counting monitor



Picture. 7.2.: High resolution spectrometry monitor

EUNDETRAF

Chapter 8

Dismantling techniques

Chapter summary

8.1. Thermal cutting techniques

8.1.1. Oxygen cutting

Gas Processes

Thermal cutting techniques applying gas flames are used for flame cuttable materials. Most of the process heat is released during the exothermal combustion of the workpiece. The remainder part is released by burning the heating gas in oxygen. The resulting flame heats the workpiece to its ignition and oxygen is then added in which the workpiece is burn.

Two different kinds of fuel can be used to provide the heating flame:

- fuel gases, such as hydrocarbon compounds and hydrogen, and
- liquid fuels.

Liquid fuels are more difficult to handle and less effective compared to hydrocarbons and should only be used for exception cases in underwater cutting.

For safety reason, only hydrocarbon compounds should be used in nuclear facilities, since hydrogen can bring about the risk of simultaneous oxygen and hydrogen gas formation.
The most important parameters for the selection of fuel gases are the maximum flame temperature, the primary flame efficiency and the density of the gas in comparison with the density of air (approx. 1.2 kg/m^3).

Hydrocarbon compounds data

Type of fuel	Max. flame temperature (°C)	Density (kg/m^3)
Acetylene (C_2H_2)	3150	1.17
Propane (C_3H_8)	2828	2.02
Methane (CH_4)	2786	0.72
Natural gas	(values depend on the composition, mainly CH_4)	

Flame cutting

Flame cutting is a thermal cutting technique which is performed with an oxy-fuel gas flame and cutting oxygen. The principle of flame cutting is shown in Figure 1.
The heating flame heats the workpiece locally to its ignition temperature. The flame then keeps the workpiece at this temperature during the cutting process. In this way, it compensates for heat dissipation by conduction into the workpiece and the environment. After the ignition temperature is reached, cutting oxygen jet is added. In most cases, this jet is located in the centre of the preheating flame to ensure successful cutting in any direction.

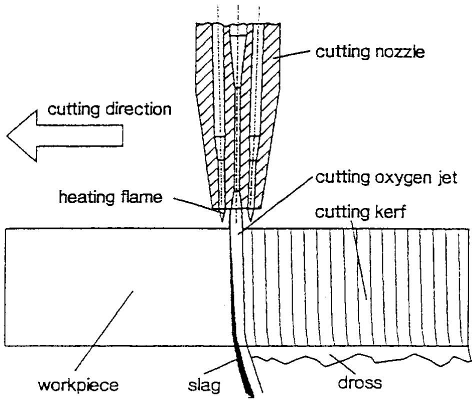


Figure 1 – Schematic representation of flame cutting

The cutting oxygen jet basically has two functions: First, it consists of oxidizing the workpiece material during an exothermal combustion process and second consists of the use of the kinetic energy of the jet to blow out the molten products containing oxides generated during the process. The cutting kerf is formed by the relative motion between the cutting jet and the workpiece.

In principle, the material to be cut has to fulfil the requirements of a flame cutting process, which are the following:

(1) The material must react with oxygen in an exothermal combustion process.
(2) The ignition temperature of the material must be lower than its melting temperature. For temperatures above the ignition temperature, the combustion heat exceeds the dissipated heat. For mild steel, which is well-suited for flame cutting, the ignition temperature is about 1150 °C.
(3) The melting temperature of the generated oxides must be lower than the melting temperature of the material to be cut.
(4) The viscosity of the slag should be as low as possible.
(5) The thermal conductivity of the material should be as low as possible.
(6) The combustion energy should be as high as possible.

Due to these restrictions, the use of flame cutting is limited to mild steel with a carbon content of up to 0,3 % can be flame cut without additional preheating. For steel with a carbon content of 1.6 to 1.8 % and higher, the ignition temperature is higher than the melting temperature. Stainless steel and the remaining non-ferrous metals are not suitable for flame cutting without additional powder injection. In the field of nuclear facility decommissioning, mainly conventional torches for flame cutting in atmosphere are in use so far.

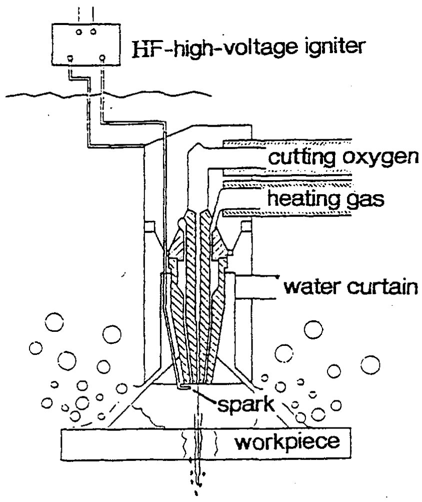


Fig. 2: Underwater flame cutting torch with a conical water curtain

Underwater use

For combination cutting processes which were developed for the dismantling of reactor pressure vessels, torches have to be modified for underwater applications.

Firstly, a special protection of the heating flame is required in a way that burning underwater is possible. This can be provided by an additional protection cap or by a concentric water curtain of a conical shape between nozzle and workpiece. The latest versions use compressed air instead of water to form this curtain.

Furthermore, the possibility of remote-controling the ignition of the heating flame underwater should be acquired. The best available igniters are high-voltage high-frequency devices, generating a spark between a separate tungsten electrode and the cutting nozzle.

Cutting Characteristics

The cutting speed mainly depends on the handling technique (manual/mechanical), the type of cutting nozzle, i.e. the design of the cutting oxygen bore, and the applied hydrocarbon fuel. Investigations in the scope of a CE project recommended propane for cutting thicknesses > 100 mm compared to acetylene because of higher cutting speeds.

Cutting speeds for flame cutting in atmosphere

	cut thickness (mm)	cutting speed (mm/min)
Mechanically guided oxyacetylene torch:	10	700
	40	460
	110	250
Manually guided oxyacetylene torch	10	370
	65	115
	110	21

Cut Thickness

In atmosphere, a maximum cut thickness of more than 2000 mm can be achieved for mild steel.
Underwater, the cut thicknesses obtained whith flame cutting range between 3 and 450 mm for mild steel.

Oxygen Lance Cutting

Two different types of oxygen lances are available:

- packed lances and
- powder lances.

An oxygen lance consists of a tube into which a core wire is inserted. Both, tube and wire, are made of flame cuttable mild steel. The principle of the packed cutting process is a follows (Figure 3).

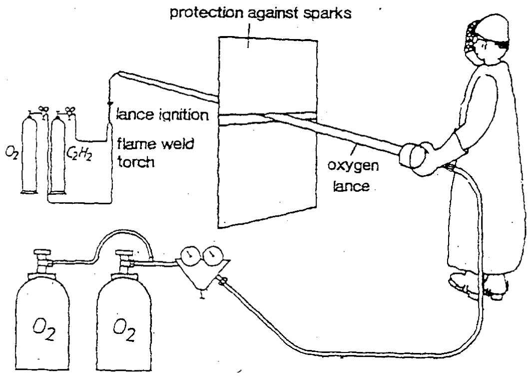


Fig. 3: Equipment for oxygen packed lance cutting

The free-end of the oxygen lance is heated up to its ignition temperature by means of an external heating flame and then burnt in the pressurized oxygen which is fed through the lance. The temperature reaches values of up to 2500 °C and the emitted heat melts the metallic workpieces as well as some mineral materials. The molten material is then blown away by the oxygen jet.

The optimum ratio of the cross sectional area of the enclosing tube and the wires and the oxygen stream area is 1.2 : 1 : 0.7. Typical packed lances have an outer diameter of 3/8". For cutting of ferritic steel, the lance may be considerably smaller. The reason for this is that a smaller amount of heat to provide the ignition temperatures is required hence only a carrier for the oxygen is needed.
In the case of oxygen powder lances, metal powder is blown through a hollow tube by means of compressed air (Figure 4). A mixture of iron and aluminium powder is normally used. The exothermal combustion of the powder increases the dissipating heat as well as the kinetic energy of the jet for blowing out the molten and/or oxidized material. Thus, the temperature increases from 2000 °C to above 4000 °C.

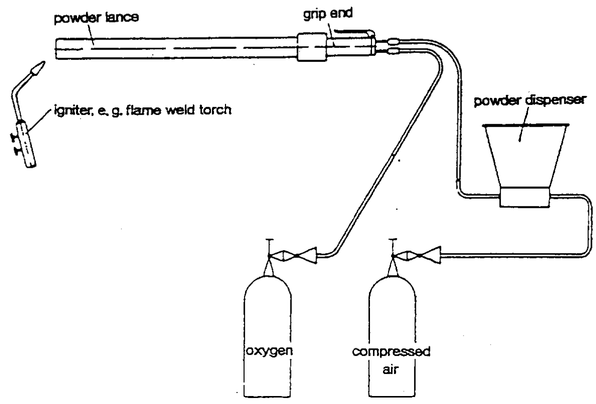


Fig. 4: Equipment for oxygen powder lance cutting

Oxygen lance cutting with packed and powder lances is used for cutting high-melting metallic material and mineral materials such as concrete. Mild-steel is burned in the oxygen jet, concrete and stainless steel are fused and blown out of the hole by means of the oxygen jet and the combustion gases.

Oxygen lances are usually guided manually. Automation of the process would be difficult to maintain and control, and therefore extremely costly. For severance cutting of metals, the lance has to be inclined with respect to the workpiece surface in the direction of cutting. This position is similar to the one in gravity oxy-arc cutting. In this way, a complete severance cut can be produced without additional subsequent machining. For greater cut thicknesses, the lance has to be guided like a compass saw, i.e. periodically up and down, parallel to the lance axis. In the case of cutting concrete, only a single hole in the workpiece is generated instead of a straight kerf. The so-called perforation cut which consists of a sequence of several holes in a row represents the first choice for the severance cutting of concrete. The remaining material must be removed mechanically.

Underwater use

The underwater use of oxygen lances is possible and has already been successfully carried out for off-shore applications. However, this technique is not widely used underwater for cutting steel less than 40 mm thick.

Following the penetration of the lance into the workpiece, there is an atmosphere-like condition around the hole, since the intrusion of water into the hole is impeded by the emerging oxygen, the combustion gases and the slag. For cutting in atmosphere, the lance must point upwards, so that the molten metal can drain off. In contrast, for underwater use, the lance must be pointed downwards in order to enable the emerging gas bubbles to rise. The area ratio should be 0.6 : 1 : 0.35 (cross sectional area of enclosing : wire : oxygen stream areas) for cutting metals and water and 0.9 : 1 : 0.5 for cutting concrete under water. In comparison to cutting in atmosphere, the oxygen consumption can be reduced by half.

Cutting Characteristics
Typical packed lances have an outer diameter of $^3/8$".

Cutting Speed:
The drilling speed that can be achieved is dependent on the workpiece material.

Cut Thickness:
The thickness of concrete suitable for drilling depends on the cutting position: Horizontal approximately 2 m, vertical – top to bottom approximately 1.5 m.

Cutting data for oxygen lance cutting

Material	**Feed rate (mm/min)**	**Consumption of 02 (1/min)**	**Hole diameter (mm)**	**Consumption of Lance (mlance/mhole)**
In atmosphere:				
concrete	200	500	45	4.25
mild steel	225	550	40	7
under water:				
mild steel	200	400	45	4-6

8.1.2 Plasma Arc Cutting

Plasma Arc Process

Thermal plasma is an elctrically conductive gas or gaseous mixture mainly consisting of ions, electrons and neutral atoms. Monoatomic gases such as argon and helium, polyatomic gases such as nitrogen and hydrogen and also mixtures of these or air can be used as plasma gases. Prior to ionisation, the polyatomic gases have to be dissociated. The plasma arc is constricted by means of a copper nozzle: The thermal and electric pinch effects are used to attain temperatures which are considerably higher than the ones obtained with open arcs described in the section above. The maximum temperature in the inner plasma arc is approximately 20.000 K and higher.

There are two groups of plasma arc processes:

- Those with a transferred arc (the arc strikes between the electrode and theworkpiece) and
- those with a non-transferred arc (the arc strikes between the electrode and the nozzle).

For practical applications, the transferred arc is almost exclusively used for cutting and eroding any conductive material. The nontransferred arc is able to cut any material, i.e. also non-conductive materials, but significantly less energy is transmitted to the workpiece.

Plasma Arc Cutting with a Single Torch

Conventional plasma arc cutting with a transferred arc is a pure fusion cutting process by which any conductive material can be cut. The plasma arc, with its high energy density, melts and partially evaporates the workpiece. The high kinetic energy gas jet blows the molten material out of the kerf. In cutting processes with direct current, the torch electrode forms the cathode and the workpiece forms the anode. The amount of heat transferred to the workpiece mainly consists of energy resulting from impacting electrons, the Joule effect and convection. A high-frequency, high voltage is uses for ignition of the arc between the electrode and the nozzle, which is connected as an auxiliary anode. A low-energy pilot arc is generated by a spark which jumps between the electrode and the nozzle. This arc is then forced into the direction of the workpiece positioned underneath. After the arc strikes the workpiece, the full cutting current can be applied and the nozzle is disconnected.

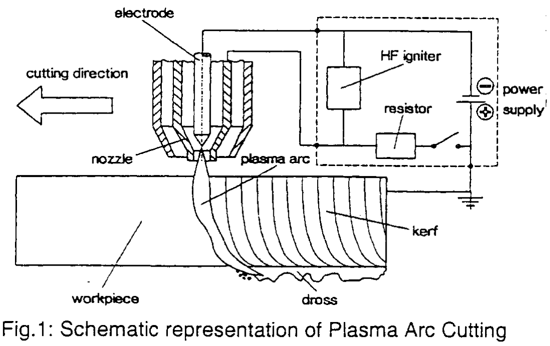


Fig.1: Schematic representation of Plasma Arc Cutting

The following techniques can be grouped according to the plasma gas used:

Argon/Nitrogen/Hydrogen-Technique

For this cutting technique, a mixture of argon, nitrogen and hydrogen is used as a plasma gas. Depending on the cutting task, the proportions of these gases can be varied and occasionally even gaseous mixtures without hydrogen and/or nitrogen are used. The cutting torch has a point tungsten electrode which may be doped with thorium oxide or with lanthanum oxide. Cooling of the copper nozzle is often applied only indirectly. Without an additional shielding gas, this variant is mainly used for cutting in atmosphere, for which currents of up to 1000 A can be reached.

Dual-Flow Technique

The dual-flow technique is a further development of the $Ar/N_2/H_2$ technique and is especially suited for underwater cutting. Here, the plasma arc is surrounded by a shielding gas, such as carbon dioxide or compressed air. As a result of this measure, the plasma arc is protected against surrounding water, energy losses are decreased and the cutting quality for mild steel is increased.

Compressed Air Technique

In this process, compressed air is used as a plasma gas. Since oxygen is present, special electrode materials such as zirconium oxide or hafnium oxide have to be selected. These materials are inserted into directly cooled flat copper electrodes in the form of small plates. This technique is especially suited to cut mild steel under atmosphere conditions. For underwater use, an additional secondary gas e.g. compressed air, will decrease the energy losses.

Water-Injection-Plasma-Cutting (WIPS) Technique

The special feature of this type of cutting torch is the nozzle, the upper part of which consists of copper and the lower part of ceramic. Between these two parts, a concentric water curtain with a low flow rate < 2 l/min is sprinkled onto the plasma arc. This leads to a further contraction of the arc and thus to an increase in energy density and temperature. In addition, a swirl ring at the cathode leads to a rotating motion of the plasma gas around the electrode, which stabilizes the plasma arc. For current intensities of 500 A, nitrogen is used as the plasma gas, for current intensities of 250 A, oxygen is applied.

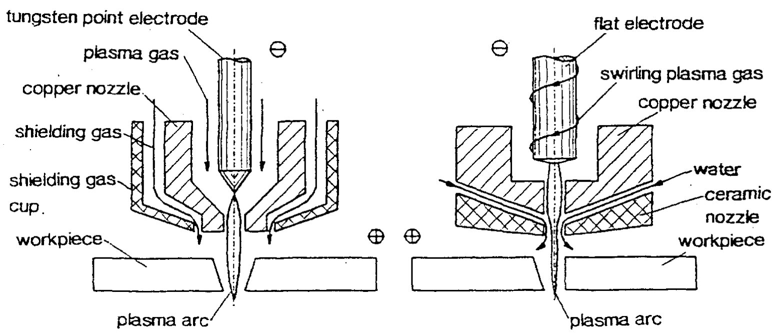


Fig. 2: Dual-flow (left) and WIPC (right) plasma arc cutting torch (schematic)

For decommissioning purposes, modular cutting torches were developed for the remote-controlled replacement of worn parts by means of manipulators. Thus, those parts with the highest wear rate, i.e. the nozzle and electrode, can easily be replaced and the torch can be adapted for individual cutting tasks. This also gives the possibility of switching between straight and cranked cutting units. Such a units has to be as small as possible, since it is used for cutting confined, complex structures.

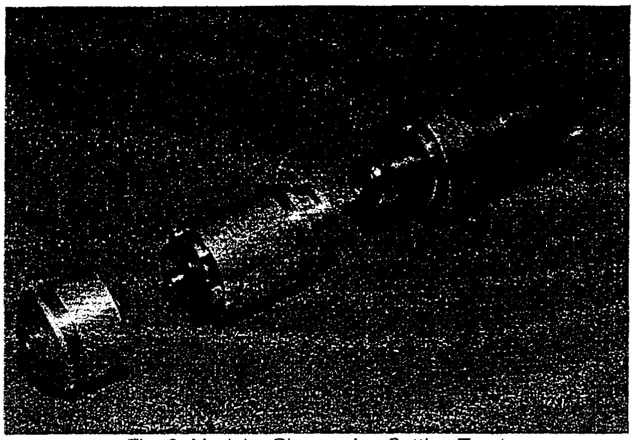


Fig. 3: Modular Plasma Arc Cutting Torch

Cutting Characteristics

The cutting speed and the cut thickness which can be obtained mainly depend on the process variant being used.

Cut Thickness

The maximum cut thickness obtainable in atmosphere is 172 mm for stainless steel, 150 mm for mild steel and 80 mm for aluminium.
The cut thicknesses obtainable underwater are 100 mm for steel.

Cutting speed for plasma arc cutting

workpiece thickness (mm)	10	20	40	80
cutting speed (mm/min)	3500	1800	1100	600

(The maximum limits for cutting stainless steel and mild steel in atmospere (I=750A))

Aerosol Emission

In order to render an effective filtration of the dust, it is important to know the size and distribution of the particles. This is even more significant when considering the fact that the filtration characteristic of high efficiency submicron particulate air filters shows a minimum for particles sizes which mainly for plasma cutting.

For aerosol measurement tests at IW, Hanover, a standard plasma cutting current of 600 A was used. Cutting parameters and gas flow for the torch were chosen in a way that severance cutting was guaranteed. Fig. 4 shows the differences between cutting underwater and cutting in atmosphere. The measured dust quantities per meter cut are illustrated for different sheet thicknesses.

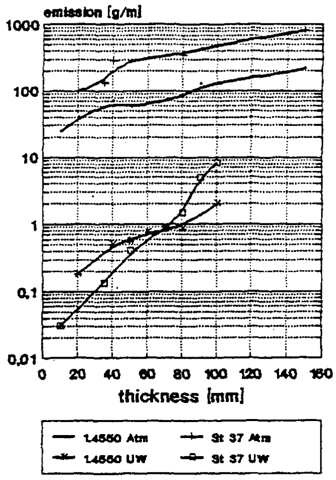


Fig. 4: Emission rates when cutting mild steel and stainless steel in atmospher and underwater

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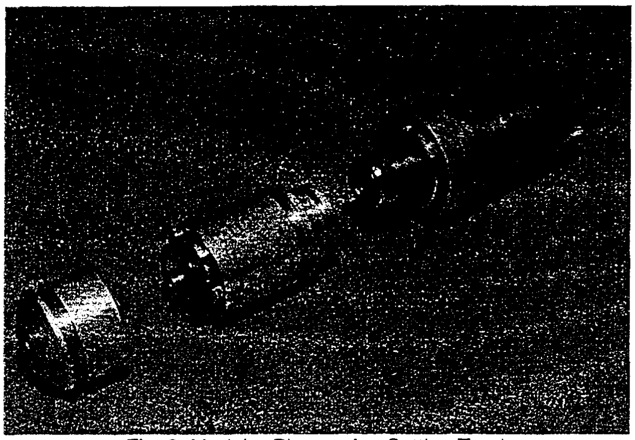

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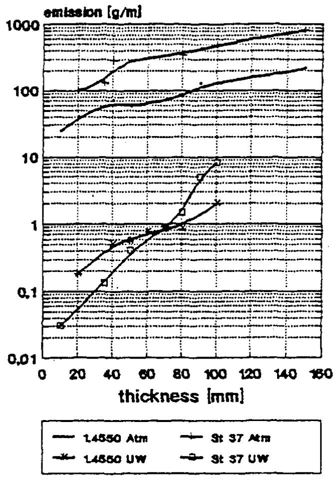


Fig. 4: Emission rates when cutting mild steel and stainless steel in atmospher and underwater

8.1.3. CAMX - A High Performance Cutting Technique for Underwater Use

Introduction

The fields of application for powerful underwater cutting techniques are manifold. Among these are applications in shallow water areas, like construction, maintenance and repair of harbour facilities, canals and vessels. Applications in deep water include offshore-installations [1] and salvageoperations. Additionally, there are also highly specialized uses, for example the decommissioning of nuclear installations. Especially the decommissioning of nuclear installations is an enormous challenge, because of material thickness and compositions. In small and radioactively contaminated rooms, the use of most of mechanical cutting tools is only possible with high constructional expense. Thermal cutting techniques offer solutions to these problems. They do work without restoring forces and, thus, guiding machines may be smaller and simpler. The most engaged thermal cutting techniques are underwater plasma arc cutting, autogenously cutting and consumable electrode water jet cutting. Mechanical-hydraulical cutting techniques complete these thermal cutting techniques [2]...[4].

During the past years, a new cutting technology was developed at the Institute of Materials Science in Hanover, the CAMX-process family (**C**ontact-**A**rc-**M**etal-**X**). Here, X symbolizes three different variants of the process, i.e. **C**utting, **G**rinding and **D**rilling. All of these variants are electro-thermal cutting techniques, which cut conductive materials with Joule and arc heating. The electrical arc flame is the result of a short mechanical contact between the cutting electrode and the workpiece. Basic components of those three technologies are one or several DC-welding transformators, powering the cutting devices with required electrical power.

Contact-Arc-Metal-Cutting

Contact-**A**rc-**M**etal-**C**utting (shortened CAMC) with a swordlike cutting-electrode is a thermal cutting technique currently used for decommissioning of nuclear facilities [5]. A waterjacketing electrode made up of pure graphite, carbon fibre reinforced graphite or a special tungsten-cooper-alloy, melts the metallic workpiece in a cyclic process by resistance heating and a free burning high current spark channel. A Master-Slave-Manipulator leads the electrode through the components, free of restoring force. Therefore, the manipulator may be designed relatively simple. With CAMC, all electrically conductive materials can be cut, including plated metal plates. The maximum component thickness that can directly be cut depends on geometry of the tool electrode and the efficiency of waterjacking, which is rinsing molten particles out of the kerf.

By this technology, complicatedly designed components like tube-in-tube-workpieces and components with re-entrant angles can be separated within a single cut. State-of-the-art of CAMC is cutting of 260 mm thick components. The kerfs show widths of 4 to 8 mm and the wastage ranges from 20 to 25%. The latest development is a computerized process control for the feed motion. This process control optimizes the feed motion depending on the resistance heating and the free burning high current spark channel. A special CAMC-tool with a turntable drive unit and an integrated process control for automatically cutting was developed by scientists of the Institute of Material Science of Hanover.

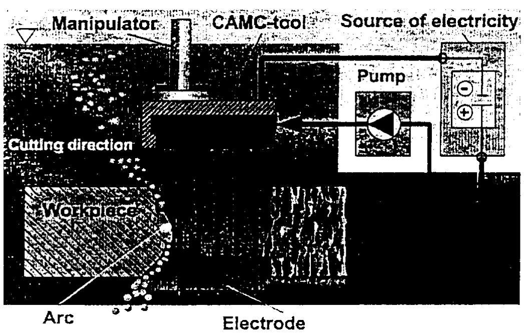


Fig. 1: CAMC-principle

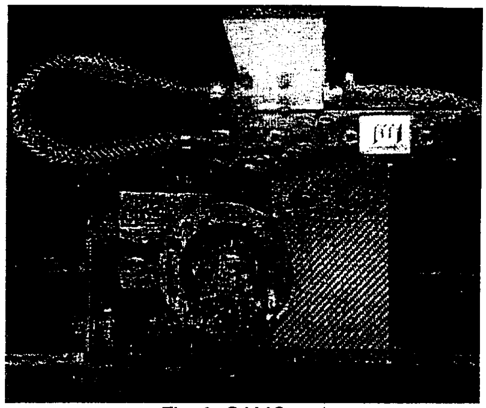
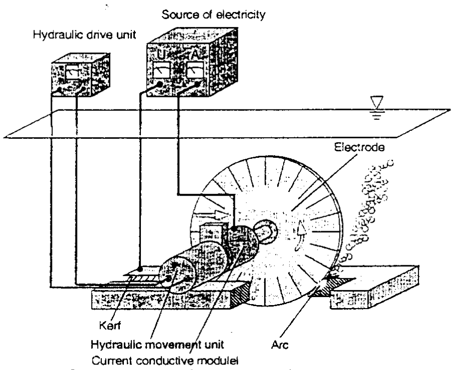


Fig. 2: CAMC-tool

Contact-Arc-Metal-Grinding

Another cutting technique is the **C**ontact-**A**rc-**M**etal-**G**rinding (CAMG) with a rotating electrode, offering new fields of application. Here, electric power is transmitted by a special mercury filled chamber within the tool. The maximum electrical power of this tool is 275 kW (5.000 A / 55 V). As materials for the cutting electrode, steel or carbon fibre reinforced graphite can be used. Due to the substantially smaller costs, combined with comparable cutting performances, steel electrodes are preferred. On principle, there are no limitations of CAMC and CAMG concerning the capability of cutting complicated structures of workpieces.
The cutting speed is very high: For example, CAMG is capable to cut workpieces of 15 mm thickness at a speed of 3 m/min. This can't be reached by any other thermal underwater-cutting technique. The wear of the rotating electrode can be reduced to 9% by appropriate parameter adjustments, and the maximum cutting thickness is 40-50 mm. In the next years, it will be possible to cut thicknesses of more than 50 mm.



Power Source statt Source of Electricity
Fig. 3: CAMG-principle

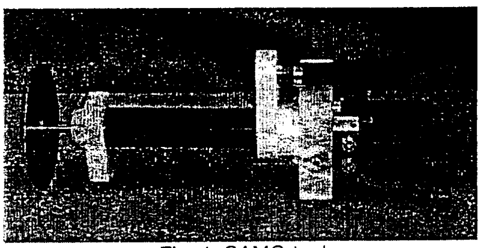


Fig. 4: CAMG-tool

Contact-Arc-Metal-Drilling

The last of the three CAMX-processes is Contact-Arc-Metal-Drilling (CAMD), which was also developed by the Institute of Materials Science in Hanover. CAMD is a novel technology to drill holes or pocket holes without restoring forces. In the CAMD-tool, there is a warp mechanism to fix and carry the workpiece. At first, a rectangular countersinking graphite electrode produces a drill hole. After that, the drilling mechanism make a turn of 90° and a hydraulic warp mechanism fixes the workpiece. Subsequently, the workpiece and the drilling tool can be carried away.

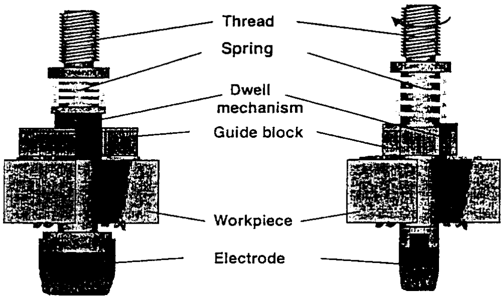


Fig. 5: CAMD-principle

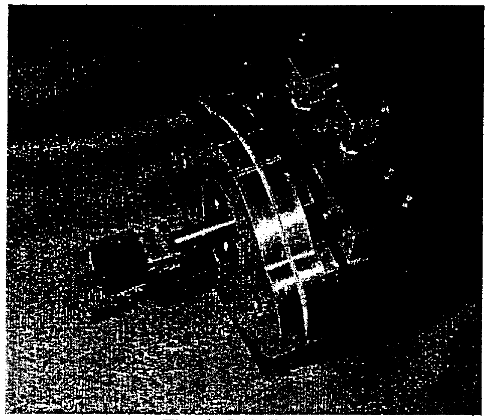


Fig. 6: CAMD-tool

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8.1.4. Basics of laser beam cutting for the dismantling of nuclear power plants

8.1.4.1 Introduction

Laser beam cutting offers the possibility to cut steels with thicknesses up to 20 mm economically by using lamp-pumped Nd:YAG lasers with average output powers up to 4 kW. Recently, average laser output powers up to 6 kW could be made available for industrial applications. Moreover, the efficiency of solid-state lasers could be increased significantly due to the introduction of diode pumped solid-state lasers. Cutting of thicker workpieces in acceptable time will be possible in the future.

A laser beam cutting system consists of the components laser source, beam delivery system, processing head, which adopts the function of forming the laser beam and the gas flow with the help of a gas nozzle, as well as a handling system and controls. Here, the gas beam with its mechanical impact is responsible for the melt expulsion. The focussed laser beam only serves to heat the material. Essential parameters of the laser beam are the power density distribution, the beam spot and the beam parameter product. Moreover, the distance of the nozzle to the workpiece is of great importance for the cutting result, since flow ratio in the cutting kerf is defined by this. Changes regarding the grain structure of the base material can arise due to laser material processing. This heat affected zone (HAZ) can be used as a measure of the influence of the laser.

8.1.4.2 Laser Cutting Processes

There are three basic cutting processes: laser flame cutting, laser fusion cutting and laser sublimation cutting. The application of the different processes is dependent on the material and the given demands.

With laser flame cutting, the additional exothermal reaction energy of the burning process is exploited. Thus, high cutting speeds are available as the release of energy during the burning process can be as high as the energy input of the laser beam. The process gas oxygen is applied, which starts the exothermal reaction. The material burns to a fluid slag, which drains off easily. Therefore, the gas pressure is set to lower values compared to laser fusion cutting. Besides, the exothermal energy has to be limited in order to guarantee a controlled cutting process. At the cut edge, an oxide layer is formed by the burning reaction. This layer has only a weak connection to the workpiece.

Laser fusion cutting is characterised by the use of an inert process gas, i.e. nitrogen or argon. An additional exothermal reaction is prevented by the atmosphere of the shielding gas. Accordingly, the process works only with the input of laser power and, therefore, the cutting speed is substantially lower than it is with laser flame cutting. In return, the cut edge is not oxidised after the processing. Further treatment of the workpiece is thus not necessary. However, the fusion resulting during the treatment is viscous. For that reason, the gas pressure has to be very high in order to expel the melt from the kerf and to realise a clean and burr free cut.

High power densities are required for the laser sublimation cutting in order to transform the material directly from the solid to the vapour phase. In addition, for this treatment an inert process gas is applied to protect the material from burning. The gas pressure used is dependent on the material which has to be processed. Because of the high vaporisation temperature of metallic materials and the limitations of available power densities, only organic or plastic materials are treated using laser sublimation cutting, while metals are mainly processed by the other two cutting methods described before [1, 2].

8.1.4.3 Process Parameters for Laser Cutting

A distinction is made between active and passive process parameters where the passive parameters are fixed before the beginning of the treatment when setting-up the machine and can not be changed during the machining process. The active process parameters can be changed and also adjusted during the treatment. Passive process parameters, for example, are the material, the focussing optics and the laser system. Active process parameters, for example, are the feed, the laser power and the process gas pressure.
For optimised processing, it is necessary to adjust the process parameters to the material and workpiece. To adjust the position of the focus relative to the workpiece, first the position of the focus relative to the processing head has to be exactly determined. Here, it is possible to work with a focus finder. For CO_2 lasers, a fusion penetration in plexiglass can be carried out, alternatively.
Further, the nozzle has to be changed before the material processing takes place if it was damaged before. An adjustment of the nozzle to the laser beam is then necessary. The laser beam must be positioned in the centre of the nozzle to get an optimised melt expulsion without burr generation. The distance of the nozzle to the workpiece influences the gas current in the cut kerf and also the cutting result.
Depending on of the given material thickness, the focal length of the focussing lens has to be chosen correctly. The operative range in the beam is characterised by the Rayleigh length. It is defined by the distance of the focus to the cross-sectional area in the beam which is twice as big as the cross-sectional area of the focus. The treatment of large workpiece thicknesses requires a long Rayleigh length and therefore, also a long focal length. For thin workpieces, a short focal length of the lens can be chosen.
The operational mode of the laser is dependent on the desired result and the demands on the product. For laser beam cutting, pulse and superpulse operation is the best.
The maximal possible cutting speed is dependent on the available laser average output power. The higher the available maximum average output power, the higher the possible feed. The influence of the average laser power on the feed is, for a wide range, nearly linear.
Absorption of laser radiation is dependent on various factors. During keyholing, reflection characteristics play an important role. The reflection is dependent on the material which has to be treated, the wavelength of the laser radiation, the angle of incidence and the surface characteristics of the workpiece. Radiation with short wavelengths (visible light, near infrared) are absorbed better by metals than long wavelengths (middle infrared). In general, non-ferrous heavy metals are more difficult to cut than steels, because of their high reflectivity and their high capability to conduct heat. With increasing surface roughness, radiation is absorbed better. Also, coatings or oxide films contribute to an increase of absorption [3].

8.1.4.4 Use of the Laser in Dismantling Technologies

Comparison of thermal cutting technologies

Laser cutting is characterised by small cutting kerfs and precise cutting contours, small heat affected zones, small tolerances, little distortion of the workpiece, stress-free treatment and high reproducibility. On the other hand, a high investment is necessary, and the low efficiency of lasers is coupled to high energy consumption. However, compared to other thermal cutting techniques, the laser produces the smallest cut.

A considerably reduced amount of particle-shaped process emissions are released. The comparison of the laser process with the plasma process relating to the Nominal Hygienic Air Requirement Limit Value (short: NHL) shows that less fresh air is needed to meet the tolerable limit values with the laser processing than with plasma processing. The NHL-value characterises the amount of fresh air which is theoretically necessary with a given emission mass flow not to exceed the tolerable limit value for each hazardous substance in the air [4].

Components of a Laser Dismantling System

Like conventional laser material processing, for the dismantling of nuclear power plants, a system consists of the laser source, beam delivery, beam shaping devices (lenses or mirrors), gas nozzle and process gas, as well as the handling system with controls. Because of different wavelengths of different laser types, different beam delivery systems are necessary. The CO_2 laser beam can be delivered to the workpiece only by mirror systems, while there is the possibility to use fiber optics for Nd:YAG laser radiation. For that reason, for the dismantling of nuclear power plants, an Nd:YAG laser is much more flexible for the beam delivery. Beam shaping is influenced by optical components on the laser beam, such as the use of a focussing lens. Beam shaping gives a characteristic form to the beam, in order to use, for example, the small focus of a beam for material processing [5].

Use in nuclear facilities

Laser technology can be used in many areas of dismantling nuclear power plants. Experimental investigations on laser processing under water were carried out, for example. First, the process gas is switched on, in order to protect the optical components and to keep water from getting into the processing head. Afterwards, the processing head is dipped into the water where the cutting process is carried out. The water does not hinder the laser beam, as the gas beam displaces the water and guarantees the free propagation of the laser beam.

Also, in the area of asbestos cutting, the laser offers process specific advantages. The release of cancerous fibrous aerosols can be significantly reduced using the laser, as the asbestos is vaporised and condenses to harmless spherical particles in the air. At the same time, the cut edge of the asbestos material is glazed during the treatment. Thus, a durable sealing of the cut edge is guaranteed, and the release of remaining fibers from the asbestos material is prevented [5, 6].

Emission-minimised cutting using special process parameters offers the possibility to attach the molten material on the underside of the workpiece in form of a burr. This reduces the release of emission products and contaminations. Also, the cutting of tubes is possible by laser material processing. Only the material and its thickness is decisive for the process. The production of a burr is also possible in this case [7].

For the dismantling of tanks or storage basins consisting of concrete walls which are lined with steel plates, cutting of the steel material is difficult, for example. The metal sheets lie directly on the concrete, and it is very difficult to cut them mechanically. Thermal cutting methods producing very deep fusion penetration in the subjacent concrete can lead to contamination of the concrete. In that case, a further treatment of the concrete would be necessary. Laser cutting offers specific advantages here. The energy input is very precise and can be controlled in depth so that the process can be adjusted exactly to the thickness of the workpiece. Fusion penetration in concrete can therefore be minimised. Moreover, there is the problem of the treatment of coated sheets. The sheets can be processed nearly without burning the coating. An additional advantage of this method is the separation of gas and laser beam. A special nozzle technique can be used to expel the molten material to the top surface of the sheet. A specific removal by suction of the released process emissions is also possible [8].

The mobility and flexibility of the laser is an important reason for its application in nuclear facilities. A condition for the realisation of these applications is the availability of hand-held laser processing heads. For this reason, a device was developed at the LZH which allows guidance by hand. The feed rate is given by the system, to guarantee a stable cutting process. With this device, which was specially constructed for the demands of the dismantling of nuclear facilities, programming and teach-in procedures are not necessary, as for example with the use of robots, which leads to significantly lower costs and saving of time. For low contaminated areas, this system offers a useful alternative to other thermal cutting techniques providing all advantages of the laser technology [9, 10, 11].

Acknowledgement

The authors wish to thank the German Federal Ministry of Education and Research (BMBF) and the Project Body Responsible For Disposal (PTE) for having funded this research (FKZ 02 S 7818, FKZ 02 S 8102). In addition, they wish to thank Energiewerke Nord GmbH, Ingenieurbüro K+S GbR and Rofin-Sinar Laser GmbH for their good co-operation and kind support.

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8.1.5 Thermite Cutting

The mean of process of thermite cutting conclude in the heating of metal in zone of cutter and its melting, which is produced due to thermal energy, resulted from combustion of thermite mixture. The products of combustion of thermite mixture flow out of combustion chamber under pressure in form of gaseous condense mixture, which provide the cutting of practically all the metals in all space positions.

The selection of content of thermite mixture and sizes of flare cartridge is required for dismantling of concrete equipment. The rigging by remote-controlled technics is required for cutting of radioactive equipment. At the present time the pilot installation for cutting of pipe-line have been created and tested.

8.1.6 Thermo-gas-jet Cutting

The given method is based on use of method of melting of material to be distracted by means of thermal and power action of high temperature supersonic jet, arising during combustion of hydrocarbon fuel (diesel fuel, kerosene, benzene) in oxygen media, and reactive oxygen burner (ROB), which is the generator of mentioned high temperature jet. The robe is similar to the camera of burning of liquid reactive motor on principle of action and working principle (picture 3).

At the present time the demonstration installation has been developed and tested which can be used as independently as in combination with other technologies. The installation is easily rigged by manipulation devices.

8.1.7 Electro-erosion Fragmentation

The electro-erosion technology fragmentation, for example, of technological channel of reactors of type RBMK is based on combination of simultaneous use of method CAC, applied for separation of non radioactive parts from radioactive ones, and technology of electro-erosion (electro-spark) destruction of radioactive parts up to insoluble radioactive slime (in form of metallic powder) (picture 4).

Later on this radioactive metallic slime can be gathered, compacted and buried. The coefficient of compactisation of solid radioactive waste can reach 0,6-0,7, moreover the accompanied secondary solid and liquid radioactive waste are practically totally absent in this technological process.

8.2. Mechanical cutting techniques

The mechanical separation processes are the oldest separation processes in the history of mankind. Already in the stone age the mechanical separation with simple stone knifes was usual. In the antiquity metalworking had progressed already far and partly a status had been achieved, which was no more reached in the following centuries.

Only in the modern times the technique was developed further noticeably, especially in consequence of the breakthrough of new energy sources such as steam and electricity. Up to the beginning of the 20th century the mechanical separation processes remained the most common procedures. Only in the last 100 years competing thermal and chemical processes were developed.

In consideration the task separation processes the following materials are to be looked at:
- Ferrous metal
    Stainless steel
    Cast iron
- Non-ferrous metal
    Aluminium alloy
- Non-metallic materials
    Concrete structures

8.2.1 Shearing

According to its definition shearing is a mechanical separation without developing shapeless materials. It is differentiated between the following kinds of cuttings, which are assigned to shearing:

- Shear cutting
- Blade cutting
- Tearing
- Breaking

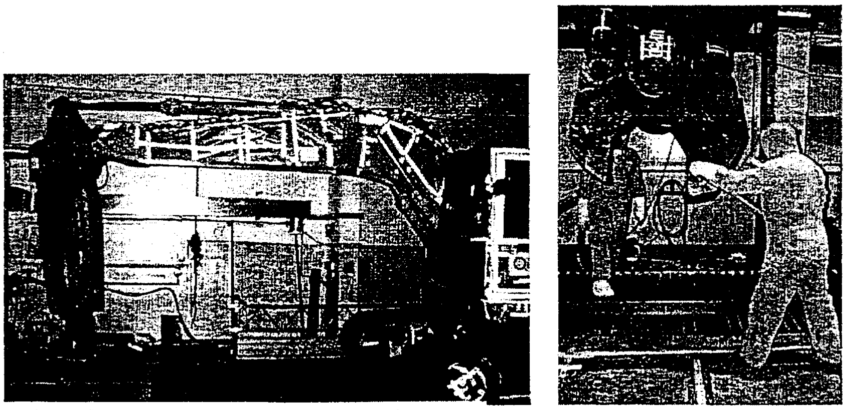

Fig.8.2.1: Excavator shears of the GNS in dismantling of the nuclear power plant "Würgassen"
For the dismantling of nuclear power plants for example hydraulic shears are used (Fig.8.2.1).

One application of this technique in the field dismantling is the fractionalization of graphitic reactor components.

Some 900 t of graphitic reactor components (moderators, reflectors and thermal columns fabricated from graphite or carbonstone) in Germany will sooner or later be subject to decommissioning and/or dismantling.
To fractionalize graphitic components, cutting techniques had to be developed and/or adapted to the special recommendations in nuclear industry: A minimization of secondary waste emission is as well desired as the ability to be applied by remote handling and as necessary as a certain performance in cutting thicker structures. Available experiences are rare and mainly stemming from former cutting operations [1].
Fractionalizing graphitic components will reduce the volume to be stored, minimize disposal costs and reach a separation of activated/contaminated and non-radioactive parts. For example, the amount of MOSAIK type II containers needed for graphitic parts of the AVR-reactor was estimated to a number of 2,511 and might be reduced down to some 50% [2].
Under radiological load, each material is subject to change of its mechanical properties, and these changes in graphitic parts are substantial to this project. As the best known radiological effect on graphite, the Wigner-Energy, can easily be released through a relatively short heat treatment, it is not considered to be an issue of scientific interest. This to an even greater intent, as lots of information from theoretical and practical experience is available.
It is well known that mechanical treatment of any kind of graphite results in dust emission from the working place. For reasons of safety, this dust has to be collected during cutting operations applied to nuclear parts. Thus, another scientific goal was reached by the development of a matching filter technique to collect graphite dust emitted by the cutting process. It was applied during evaluation works on irradiated graphite.

Graphite dust can, as every inflammable substance, cause explosions, provided it occurs in a matching environment and an ignition source as well as a fitting oxidizing agent are present. To prevent such incidences, explosion experiments with non-irradiated graphite dust were carried out to determine explosion levels. Results imply that under normal conditions graphite may explode and is unlikely to do so, if the graphite's concentration is below 250 g / m³.

Several cutting techniques (thermal, hydraulic ...) were examined, but these performed in way not suitable for the referring graphitic parts.
Technique of choice in this case is the mechanical breaking of graphitic components by means of a straddling tool as it is widely in use by fire brigades. Two arms of the tool are inserted into a hole of the graphitic part and then, by means of an electrohydraulic pump, opened (Fig. 8.2.2).
Within seconds after inserting of the tool, the graphite piece is torn mainly into two parts, depending on the geometry of the related part and the position of the arms. Another advantage of this technique lies in the negligible emission of dust out of the cleavage area. Just approx. 0.8 g is the total amount of graphitic dust taken out of a broken area as large as this sheet of paper.
Typical cutting results are shown in figure 8.2.3.

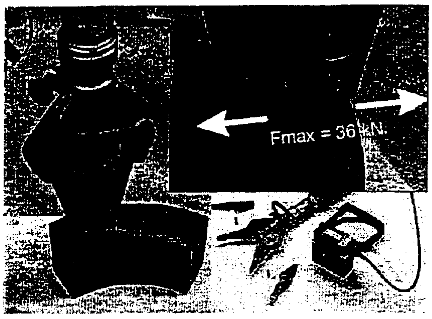

Fig. 8.2.2: Mechanical breaking of graphite

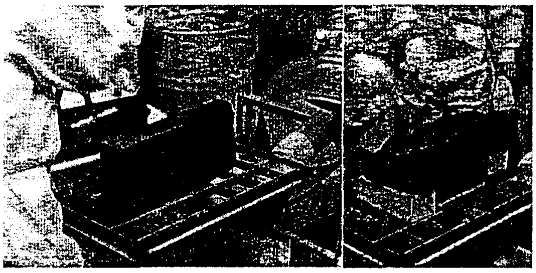

Fig. 8.2.3: Typical cutting results from mechanical cutting of graphitic components from RFR reactor

8.2.2 Sawing

According to its definition sawing is cutting with a multi tooth tool of small kerf width. With all sawing types the tool is moved and supported by a feed motion. Differences exist in performance data, in the wear of tools and in the accumulation of secondary wastes:

Fret saw:
- Cutting depths up to 100mm
- Wear of tool rises with the cutting depth superproportionally
- Tool works without coolants and lubricants in most applications

Bow saw:
- Also for thin-walled components suitable
- High tool life circles lives by characteristic movement
- Coolants and lubricants increase tool life circles
- Handling of components with dimensions to 1m cut lengths usually

Band saw:
- Handling of larger diameters
- High tool service lives by few load of the individual tooth
- Coolants and lubricants increase tool life circles
- Small kerf widths and a low amount of secondary waste can be obtained by narrow dimensions of the tool
- Very good results obtained in various decommissioning projects (as well for cutting in air as under water).

Circular saw:
- Cutting depths in metals up to 200mm
- Cutting depths in concrete up to 550mm
- Coolants and lubricants recommendable
- The use of remotely operated underwater circular sawing is currently common in decommissioning projects.

Core saw:
- Separation process is a mixture of sawing and grinding
- Wire-cable with cutting elements of boric nitride or diamond
- Coolants and lubricants are needed
- Secondary waste is predominantly powder or slurry
- Cutting depths in metals up to 300mm
- Cutting depths in concrete to 1000mm

In general, one can say that sawing is a proved industrial technique which produces few secondary waste (chips) easily collectable. It has been used successfully in different decommissioning projects worldwide.

8.2.3 Grinding (abrasive cutting)

With abrasive cuttings beside the workpiece also the tool material is removed away. The tool material is used as aluminium oxide, silicon carbide, boric nitride or diamond merged in resin and partially strengthened by fiber glass.

A cutting depth in metal up to 30mm with mobile devices has been achieved. During cutting a good heat dissipation and a stable tool guidance are necessary.

8.2.4 Blasting

A separation with blast technique is used due to its increasing economic advantages in the field of dismantling activities. The costs of a blasting-technical dismantling amount only to 10 to 20 % of the costs of other dismantling processes.
For high buildings starting from approx 50m height (for example chimneys of power plants, figures 8.2.4-8.2.7) often blasting represents the only effective method of the dismantling.

Fig. 8.2.4-8.2.7: Blasting folding of reinforced concrete chimneys of the power plant "Schwarze Pumpe"

In order to execute blast projects surely, the strategy of blasting is to be adapted and verified to the respective basic conditions. Thereby the following procedures are to be distinguished:

- Separation by blasting with preformed explosive charges
- Separation and clearing away by blasting by means of explosive charges brought into drillings
- Removing away by blasting with detonation cords

8.2.5 Milling

Milling is a procedure for the handling of workpieces with a high rate of material removal. It is not used as a separation process in the original sense. The material is processed with high expenditure of time and tools.

This procedure is not particularly suitable for dismantling of nuclear power plants and is used only in special cases (for example for the preparation of components).

Milling tools are applied in most cases as stationary units. Additionally, mobile devices are used, for example in the conventional area for milling railway switches.

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8.3 Hydraulic cutting techniques

The wide potential of jets is caused by their wide range of resulting loading regimes, from static loading to dynamic loading resulting from jet disintegration, cavitation, electrical or ultrasonical modulation. The jet technology is involved in a wide range of applications and processes besides cleaning and cutting. Examples are milling, turning, drilling, fragmentation, surface modification, etc. This results in a lot of different industrial applications like manufacturing, surface preparation, medicine, electronics, automotive industry, military, and nuclear application, for example.

The improvement of the reliability of jet cutting systems as well as the increasing implementation of automation and recycling systems of water and abrasive will rise the number of applications and accelerate the spreading into different fields of application in the future.

The increasing pressure levels of the plunger pumps, which are normally connected with higher flow rates compared to intensifier pumps, opened a wider field of effective application like shipyard cleaning, removal of concrete, nuclear application, etc.

8.3.1 Cutting and decoating with pure waterjets

The plain waterjet is generated by pressurising the water with an intensifier or plunger pump. The pressure energy is transferred with a small nozzle into kinetic energy. The water is accelerated up to several 100 m/s depending on pressure and flow rate. Today intensifiers deliver typically pressures of 400 MPa and flow rate of 4 l/min. Plunger pumps, with a higher efficiency, reach 350 MPa and a flow rate of 20 l/min and are used for cleaning purposes.
The pressure used for surface cleaning is related to the material properties. In the case of decontamination of concrete structure, typically pressures are in the range of 100-150 MPa.

A waterjet of a pressure up to 400 MPa is allowed to cut many materials like plastics, wood and other comparatively soft materials. However the range of materials, which could be cut, is remained limited. Metals, ceramics and other technical materials, are not machinable by pure waterjets. Only the addition of abrasive particles to the waterjet makes an effective cutting of technical materials possible.

8.3.2 Cutting with abrasive waterjets

By adding abrasives to the plain waterjet the efficiency of the tool is increased. Two kinds of abrasive waterjets are well known. The Abrasive Water Entrainment Jet or Abrasive Water Injection Jet (AWIJ) and the Abrasive Water Suspension Jet (AWSJ).

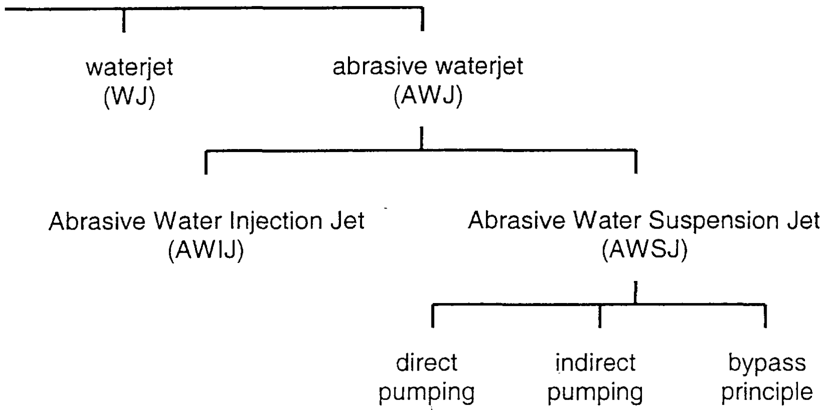
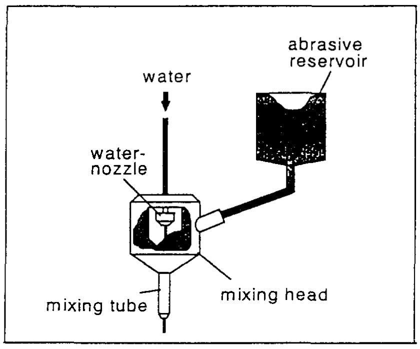


Fig 8.3.1: Division of waterjets

Abrasive Water Injection Jet (AWIJ)
The idea to generate an AWIJ was developed in the lately seventies. Main component is a mixing head, with an assembly of a water nozzle and a focusing or mixing tube. The water nozzle has a diameter of 0.2 – 0.5 mm and generates a plain waterjet. This jet runs through the mixing chamber and generates a vacuum pressure. Through an opening, the abrasive particles are sucked into the chamber pneumatically. In the mixing tube the abrasive and the water are mixed, accelerated and focused (figure 8.3.2).



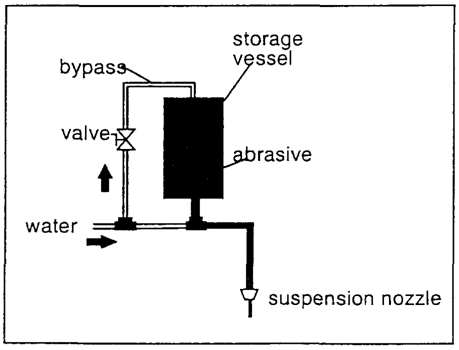


Fig. 8.3.2: Abrasive Water Injection Jet Fig. 8.3.3: Abrasive Water Suspension Jet

Abrasive Water Suspension Jet (AWSJ)
A second variation to generate an Abrasive waterjet is the AWSJ and was developed in 1984 by the BHR-Group in Cranfield, UK. A high-concentrated suspension is stored in a vessel under system pressure in the pressure circuit.
Main difference to the AWIJ is the absence of air in the jet. One part of the pressurised water is used to feed the high concentrated suspension into the main water stream. The suspension can be transported with long high pressures hoses to the cutting location (Fig. 8.3.3).
Due to the fact there is no air in the jet the efficiency of this jet is much higher than the AWIJ. AWSJs are well known in the dismantling industry and only a few applications for manufacturing purposes are known today. State of the art for AWSJs is pressures up to 200 MPa. 400 MPa-AWSJs are developed and running under laboratory conditions.

Characteristics of AWIJ and AWSJ
The characteristics of the different abrasive waterjets are based on their generation. The AWIJ consists of three phases (f.e. ≈95% vol. air, ≈4% vol. water, ≈1% vol. abrasive), the AWSJ however only consists of two phases (80-90% vol. water, 10-20% vol. abrasive). This leads to a better acceleration of the abrasive particles in an AWSJ. Therefore its cutting efficiency is at least twice as high as of an AWIJ of the same hydraulic power and abrasive flow rate (figure 8.3.4). But the AWIJ is commonly used with pressure as high as 400 MPa, which leads to similar performances as the AWIJ in terms of the depth of cut.

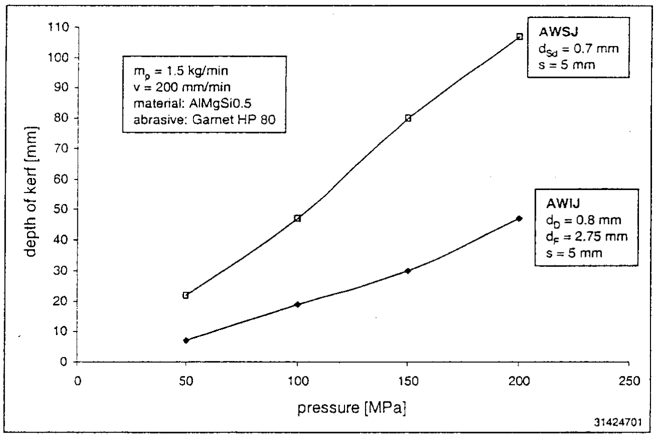
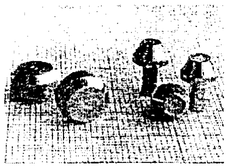
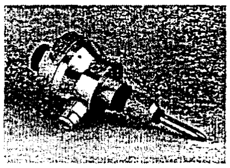
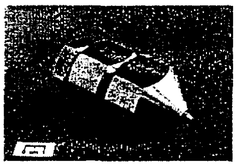


Fig. 8.3.4: Comparison of Cutting Efficiency of AWIJ and AWSJ

Due to the fact, that AWSJs only consist of water and abrasive material, the particles are guided in a better way than in AWIJ. This leads to a higher jet stability and finally to an improved cutting quality and cutting efficiency.
Generating AWIJs a waterjet leaves the water nozzle, passes the mixing chamber and then enters the focussing tube. This leads to the demand, that the focussing tube diameter is at least twice (normally 3-4 times) as large as the water nozzle diameter. Therefore AWSJs of the same nozzle diameter – and consequently the same hydraulic power- produce lower cutting widths and higher cutting depths.

The advantages of the waterjet technology in relation to thermal and conventional procedures are summarized in table 6.3.1.
Table 8.3.1: Advantages of the waterjet technologie

Principles	WJ	AWIJ	AWSJ
Multifunctional tool	cutting, drilling, turning, decoating, cleaning		
Non-thermal process	no toxically reaction products		
Omnidirectional	sharpness of the jet from every side		
Almost all materials can be cut	„soft" materials	metallic and ceramic materials	
	homogeneous- and inhomogeneous material, composite materials		
Small width of cut	> 0,1 mm	> 0,4 mm	> 0,3 mm
Achievable depth of cut	e.g. PVC 20 mm	e.g. steel 120 mm	e.g. steel 300 mm
Small and flexible tool			
Application in different environment	in air, under water, in explosive environment		
Low reaction forces	15 N – 250 N		
Low stand-off distance sensitivity	no focussing necessary		
Natural resources	water	water and abrasive	

Abrasive waterjets for the dismantling of nuclear power plants
One first example of dismantling by Abrasive Water Injection Jets (AWIJ) is the biological shield of the JPDR in Japan. Beside waterjets, diamond saw and drill, an explosive technique was used to dismantle the shield, consisting of reinforced concrete.
The advantages of the abrasive waterjet cutting technique combined with newly reached developments in combination with other advantages generated the idea to qualify this technique as an alternative decommissioning technique in the nuclear power plant (VAK)

in Kahl. Two research projects, the generation of higher working pressure as well as the application at VAK, Kahl, were sponsored by the Federal Ministry of Education, Science Research and Technology (BMBF). Within the project at the VAK, Kahl the cutting of the lower core shroud - activated material to be cut under water - and the development of a strategy to cut the reactor pressure vessel with Abrasive Water Suspension Jets were planned.

Cutting strategies for the dismantling:

Cutting through:

- For total material separation, a setting angle for cutting of 15° has to be taken.
- If other parts of the object are closed to be cut and should be affected minimal, the cutting angle can be increased up to 45° to minimize the effect of the gap.

Kerfing:

- Kerfing a definite percentage of the material thickness (for example 95%) to prevent the surroundings by impurities. The used abrasive and the machined (radioactive) material is collected inside of the pressure vessel. Only when cutting the remaining wall thickness a small percentage of abrasive and machined material are ejected to the surroundings.

First application of an Abrasive Water Suspension Jet (AWSJ) was at the nuclear power plant VAK, Kahl. At the beginning a 140 MPa cutting unit was used to cut the lower core shroud and the thermal shield. The reactor pressure vessel itself was cut by a 200 MPa unit. The data of application are given in table 8.3.2.

Table 8.3.2: Cutting parameters of VAK Kahl

	Lower Core Shroud	Thermal Shield	Reactor Pressure Vessel
Material	X 6 Cr Al 13	X 6 Cr Al 13	Austenitic plated, ferritic steel, 19 Mn 5
Material thickness	51 mm (132 mm)	32 mm	104.5 mm (6.5 + 98)
Working pressure	140 MPa	140 MPa	200 MPa
Water flow rate	8 – 20 l/min	8 – 20 l/min	9.5 – 20 l/min
Abrasive flow rate	1.3 kg/min	1 kg/min	1 kg/min
Cutting speed	40 mm/min (13 mm/min)	65 mm/min	25 mm/min
Total length of cut	20 m	70 m	63.875 m
Total consumption of abrasive	1000 kg		2553 kg

An advantage of abrasive waterjet cutting compared to thermal cutting is the small amount of aerosol, the disadvantages the secondary waste. Both are quantified and analysed for kerfing and cutting through for application in air as well as under water. Only a very small amount of waste is spread into the air as aerosols, most of the waste are sedimented particles.

To manage the waste, a catcher and a special filtering device is to be installed. At the VAK, Kahl most of the abrasive material (97%) was directly filled into a special container. Small particles were held back by a special filter system, the water could be reused.

An optimized cutting prozess and the right cutting strategy is necessary to minimize the flow rates of abrasive.

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8.4 Comparison of different dismantling cutting tools in the same experimental conditions

For decommissioning programmes, numerous studies have been carried out on cutting tools, but it is often difficult to compare the results of different studies because the tools were implemented under a wide range of conditions. The purpose of this investigation was therefore to compare various cutting tools under standard conditions of use, and thus to allow realistic assessments for consideration in decommissioning scenarios.

The comparison covers both cutting performance and secondary waste generation, with particular emphasis on the following parameters: cutting speed, secondary waste distribution (sedimented dross, cell wall deposits, aerosols) and, in some cases, characterization (particles size distribution, chemical composition).

All the tests were conducted under non-radioactive conditions in an airtight 32 m^3 cell comprising modular stainless steel panels.

The tests were conducted on stainless steel and steel specimens between 0.2 and 20 cm thick using seven types of tools: reciprocating saw, disk grinder, plasma torch (50A and 200A), arc-air cutter, arc saw, lost wire pulsed oxycutting tool (LSI) and Nd-YAG laser.

A lot of data have been obtained and interpreted and some main results can be underlined:

- For mild steel, the fastest tool was the LSI and the slowest were the reciprocating saw and the Nd-YAG laser. For stainless steel, the fastest tools were the LSI and the plasma torch (200A) and the slowest was the reciprocating saw . The aerosol production per square meter of cutting surface (length x thickness) diminishes appreciably as the surface cutting rate increases.
- The mechanical tools (reciprocating saw and disk grinder) produced a kerf practical no wider than the thickness of the blade (0.2 cm) or grinding disk (0.4 cm) and relatively independent on the workpiece thickness. The plasma torch and the laser generally produced a narrower kerf that the mechanical tools, and was also relatively unaffected by the workpiece thickness.
- The tools producing the least secondary waste (sedimented dross, wall deposits and aerosols entrained in the exhaust ventilation) for stainless steel were clearly the laser, the reciprocating saw and the 50A plasma torch while the LSI produced the most. Similarly, for mild steel the secondary waste production was lowest with the laser and highest for the arc saw and the LSI.
- Sedimented dross accounted for most (88 % to nearly 100 %) of the secondary waste production.
- The ratio of the cell wall deposit mass to the total collected mass varied from 0.07 % to 5.2 %.
- The reciprocating saw produced very few aerosols comparatively with the other tools. The LSI followed by the arc saw produced the largest amounts for stainless steel and also for mild steel followed by the arc-air cutter.
- The particle size distribution was often multimodal. The main mode was near 1 μm for the arc saw with the 0.5 cm and 5 cm cutting specimens, about 7 μm for the disk grinder with stainless steel, and near 6-7μm for the arc-air cutter with both mild and stainless steel. For the plasma torch and the LSI, the main modes were low (~ 0.1 μm and ~ 0.6 μm) for stainless steel and high (~ 6 μm) for mild steel.
- For stainless steel cutting tests, the aerosols were often enriched in manganese.
- No wear was detected on the plasma torch nozzle and for the laser. The greatest tool wear was measured for the arc saw disk.
- The production of NO and NO_2 exceeded the permissible limits during the plasma torch cutting tests especially when nitrogen was used as plasma generating gas. The carbon monoxide production reached seven times the permissible limit during the arc-air cutting tests and could reach 14 times this value during the LSI cutting tests for mild steel plates.

The aim of this study was to compare the selected cutting tools in so far as possible under the same operating and measurements conditions in the same test environment. The objective was to compile useful information in order to provide decommissioning site managers with a means for selecting the best tools for a specific task.

Moreover, the use in actual operations implies also other constraints, like the radioprotection (for maintenance purposes), the avoidance of contamination (in the air or in the water, the amount of secondary waste produced and the easiness to trap it. Those factors can lead to the selection of simpler tools or to the ones generating less secondary waste, or to the techniques currently available and used in the industry.

None of the tools may be considered the best, in as much as not all-purpose tool exists at the present time and the tools selected for one dismantling operation can be unsuitable for other. Moreover, in same countries economic or social considerations may carry greater weight than purely technical criteria.

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APPENDIX 1

STATE OF THE ART ON DEVELOPMENT OF TECHNOLOGIES FOR DISMANTLING OF NUCLEAR INSTALLATIONS IN RUSSIA

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The Russian nuclear installations, first of all that with reactors of first generation, have been projecting and created without accounting of problems of their decommissioning. In connection with this there are no specialized technologies concerns, which are necessary to use for dismantling and fragmentation of different high level radioactive constructions and equipment during decommissioning.

It is necessary to say, that on each Russian NPP the special restore services are functioning, which possess the certain issue of instruments and technological rigging which could be used for fulfilling of sufficient wide list of works, required during decommissioning of NPP unit.

Beside of this, the experience accumulated in Russia during works of restores and reconstruction of research and industrials reactors and NPP units, justify the possibility to fulfill the dismantling of equipment and systems with use of already existing technologies. Indeed, in this case, the collective irradiated dose of personnel fulfilling these works could be non reasonable high. In connection with this in Russia an issue of experimental works have been fulfilled oriented towards searching for more modern effective technologies for decommissioning of nuclear installations.

1. Development of remote-controlled techniques for dismantling of high level radioactive constructions of nuclear reactors

As the foreign and national experience show, the main directions in technology dismantling of nuclear reactors of channel and vessel types is the process of fragmentation of reactor on parts directly in the shaft of reactor.

For realization of this technology the necessity is recognized to develop the means of technological rigging, which are specialized on construction and functional possibilities-the remote-controlled of dismantling complexes (RCC), realizing the principle of implementation of "desolate" technology of production of works.

At the present time in Russia the constructors developments have been carried out of a few variants of such complexes, designated for dismantling of high level radioactive constructions of channel and vessel nuclear reactors. The principally non solved problems for development and preparation of RCC at the present time do not exist, nevertheless the complex of scientific research and pilot-construction works has to be fulfilled for their practical realization.

- ***The remote-controlled complexes (RCC) for dismantling of nuclear installations with reactors of channel type***

The dismantling of unit with nuclear reactor of channel type AMB-100 on Beloyarskaya NPP the draft project of construction of RCC have been developed (Fig.1) /1/. This RCC

presents the complicate production complex with different extent of interaction of its follow systems and subsystems:
the supporting-transport system is the power construction prepared from carbonic steel. This system is designated for location of mechanisms, knots and metal-constructions of other systems and subsystems of RCC. Beside of this the supporting-transport system fulfills the functions on localization of working zone, radiation safety of space, which is proximate with working zone, on provision of functioning of technological working organs, as well as for displacement of RCC above the working zone;
the manipulation system of RCC is designated for remote fulfilling of technological operation of dismantling;

the lifting-transport system of RCC is designated for dislocation of products of dismantling from working zone to zone of initial management with radioactive waste;

the system of initial management with radioactive waste fulfills preliminary operations on its preparation for next displacing to subdivision of treatment of arising waste;

the system of ventilation, dust suppression and filtration is designated for removal, dust suppression and filtration of secondary gaseous and dust products, arising during dismantling works;

the system of energy provision supply technological and additive equipment and mechanisms of RCC by energy carriers: electro-energy, compressed air, water, oxygen, acetylene;

the system of control of RCC is designated for control of systems and technological rigging of complex in remote and automatical regimes.

The developed construction of RCC principally solves the questions of radiation safety and environmental conservation during production of dismantling works on the reactor. In this goal the metal-construction of RCC represent the construction of closed type, which localize the working space, and divide it on three working zones with different levels of radioactivity.

It is necessary to notice that location of the most of drives of mechanisms of RCC is fulfilled behind radiation barriers and in the cases when the located in working zone (for example, near the manipulators) their construction is made from radiation resistant material.

The main technical characteristics of RCC are as follow:

overall sizes	18 000 x 15 800 x 12 000 mm
lifting capacity	
- light manipulator	350H
- hard manipulator	2 000H
lifting capacity of hoist	7 000H
masse	420 000 kg

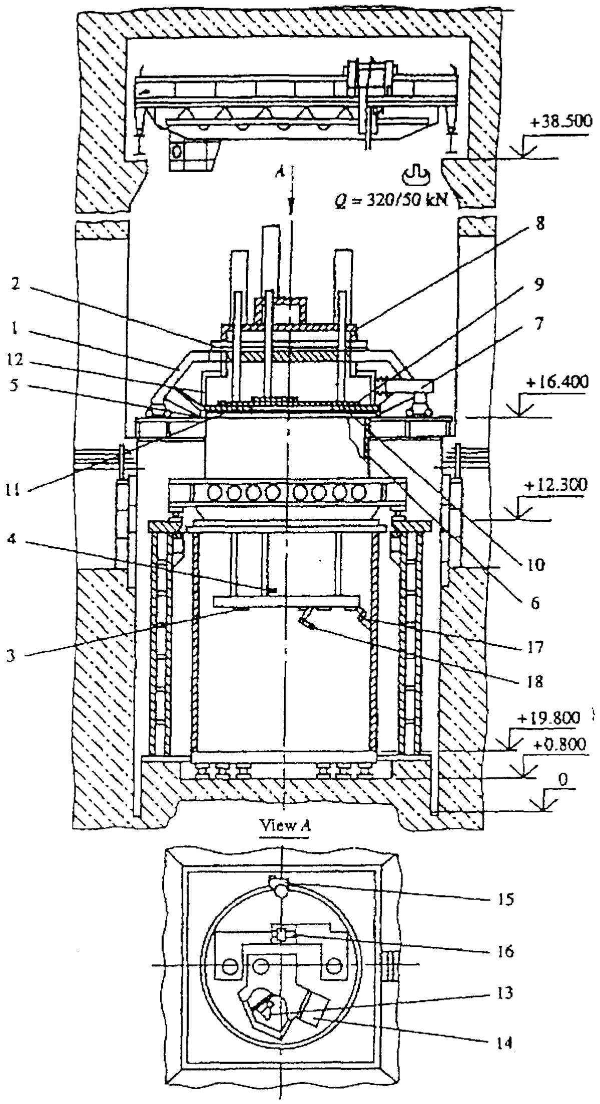


Fig. 1. The general construction of the remote-control system for dismantling of reactor of the first power unit at the Beloyarsk NPP:

1 - Bearing truss;
2 - Skewing platform; 3 - Crosspiece;
4 - Hoist;
5 - Water seal;
6,12 - Safety compartments;
7 - Repair box;
8 - Handling box;
9-11 - Radiation-safety plates;
13 - Facility for sealing and decontaminating the primary packing of radioactive waste;
14 - Auxiliary safety container;
15 - Platform turning mechanism;
16 - Crosspiece lifting mechanism;
17 - Light-duty (tool) manipulator;
18 - Heavy-duty (load) manipulator

- ***Remote-controlled complexes for dismantling of nuclear reactors of vessel type***

It is necessary to say that in principle the described above RCC although may be used

for dismantling of vessels and intra-vessel equipment with nuclear reactors of vessel type. Nevertheless in Russia with accounting of mentioned above requirements the analogous development of construction of RCC have been fulfilled for nuclear reactors of vessel type (in particular, VVER).

The construction of RCC for vessel reactors has the structure, which is analogous to that, describe above, as well as remote and automatical regimes of service of working organs, control and management of dismantling process.

In one of proposed concrete technical solution for increasing of efficiency and safety of dismantling of vessels of reactors the process of continuous milling inside of vessel of reactor is used which is realized from top to bottom on circular trajectory with offset of cuttings to the bottom of vessel of reactor throw specially foresight internal cavity of supporting rod, which simultaneously fix the vessel of reactor (Fig.2)

The method of dressing of vessel by milling is practically not accompanied by generation of secondary radioactive waste in form of gas-aerosol fractions, and that is why the use of special systems of ventilation is not required in the working zone.

From estimation, depending of mean of milling the "clean" time of dressing of reactor vessel of type VVER will be since 5100 up to 6200 hours. The expediency of use of proposed technology for dismantling of vessel reactors is determined by follow factors:

1. The construction materials from which the components of reactors have been prepared have the acceptable level of processability by cutting.
2. The construction of vessel and intra-vessel devices have simple geometrical forms and sufficient firmness what is convenient from point of view of their treatment by cutting, simplicity of kinematics of process and its control.
3. The use of concrete shaft of reactor as a localized working zone of dismantling allow to exclude the necessity of organization for this goal a new radiation-contaminated areas inside and outside of building of the unit.

The process of mechanical treatment by cutting as well as worked out constructions of instrument, technological rigging and equipment have wide industrial implementation.

4. The cuttings resulted from milling of vessel has a small length and regular form and that is why it is technologic from point of view of its next management as radioactive waste.

During milling by infusible instrument the use of cooling liquid is not obligatory what allow to avoid the arising of secondary accompanied radioactive waste.

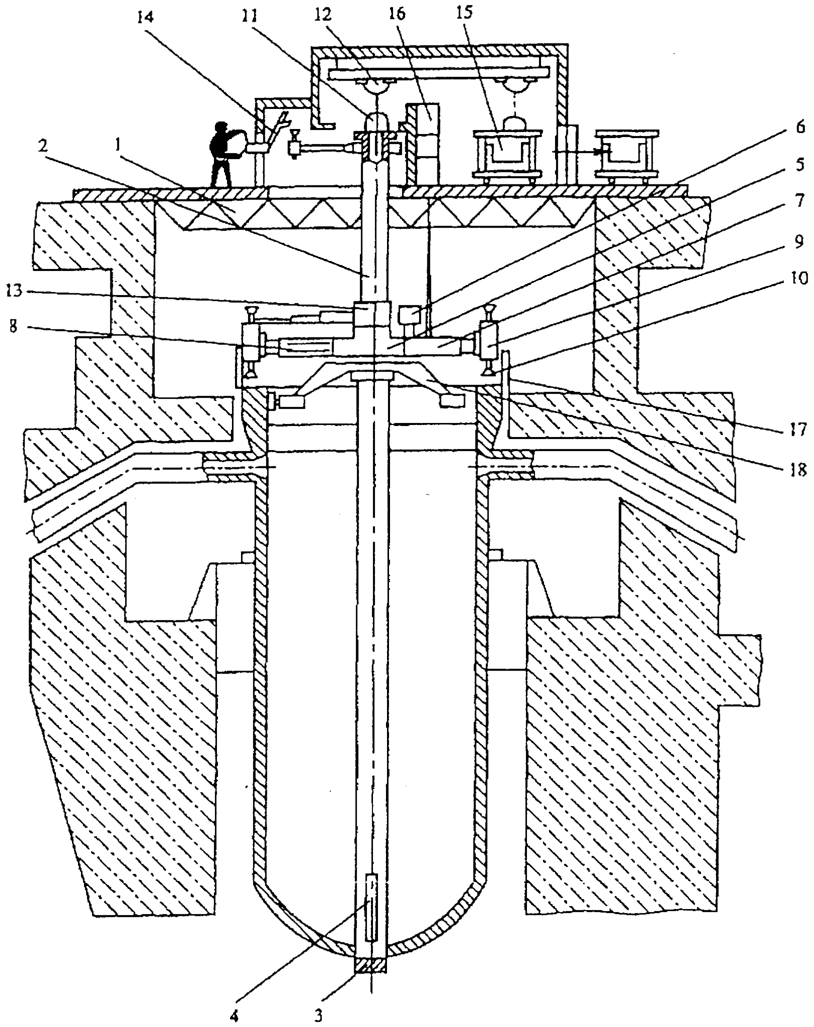
5. The process of arising of cuttings of pure metal of constructions of reactor practically has no dust and gaseous removal, what allow to make free the works from use of system of special ventilation and keep possibility of control for the working zone.
6. The construction of metal-cutting complex does not exceed on complicity the construction of other dismantling complexes for fragmentation by other means, when the use of complicate manipulators, lifting devices, systems supply of energy carriers etc. is required.

It is necessary to notice that the proposed technology is also preferable from position of provision of requirements of radiation and ecological safety of carrying out of such works.

The disadvantage of proposed technology of dismantling deal with requirement of increased energy and time consumption and comparison with dressing on fragments by size 200x200 mm and more. Nevertheless this disadvantage is partially compensated during operations of management with radioactive products of dismantling, arising in form of cuttings.

The main characteristics of construction of complex for the variant of milling, applicable to the vessels of nuclear reactors of type VVER are follow:

overall sizes:	
-height	11 000 mm
-width	14 000 mm
-length (to direction)	10 000 mm
masse (without radiation shield)	90 t
volume of obtained cuttings	400 m^3
amount of required package (barrels of capacity 0,2 m^3) under condition of pressing of cuttings	1 000 pieces
energy consumption for milling of reactor's vessel	79 000 kW x hour



7.

Fig. 2. The dismantling system for mechanical cutting of the vessel of VVER-440 reactor:

1 - Supporting structure; 2 - Pull-rod; 3 - Tip; 4 - Ports; 5 - Platform; 6 - Turning gear; 7 - Rotary table; 8 - Radial slots of rotary table; 9 - Rotary tool heads; 10 - Cutting tool; 11 - Grip; 12 - Hoist; 13 - Manipulator; 14 - Tool box; 15 - Handling facility; 16 - Platform hoisting mechanism; 17- Safety shell; 18 - Centering device

2. Development of methods of fragmentation of constructions of nuclear installations

At the present time in Russia practice the thermal methods of cutting are widely used for the cutting of metal-construction of nuclear installations, among which the contact-arc and plasma cutting are more widespread.

Beside mentioned thermal methods for fragmentation of metal-construction of nuclear installations in Russia other methods have been tested and recommended at the recent time in connection with conversion of defense enterprises of national industry /2,3/.

- ***The method of contact-arc cutting (CAC)***

This method have been developed in details and is widely used on practice. The method of CAC is based on thermal action of impulse arc charges on treated material. The given method is characterized by negligible power effect on detail to be cut, by stability of process in air and by acceptable stability in the water. It is applied for cutting of practically all construction materials (from carbonic steels, austenitic steels, aluminium) and its different combination in one construction. The different marks of graphite are used as cutting instrument of CAC.

- ***The method of plasma cutting***

The given method have some advantages in comparison with method of CAC, among which there are: high rate of cutting, better clean of surface of cutter, convenience of use on restricted working places, less amount of melting metal in process of cutting etc.

It is necessary to say that the method of plasma cutting have been widely used during dismantling and reconstruction of nuclear installations for fulfillment of different operations. In connection with this in Russia there is the wide production base on issue of required plasmotrons.

- ***The method of thermite cutting***

The mean of process of thermite cutting conclude in the heating of metal in zone of cutter and its melting, which is produced due to thermal energy, resulted from combustion of thermite mixture. The products of combustion of thermite mixture flow out of combustion chamber under pressure in form of gaseous condense mixture, which provide the cutting of practically all the metals in all space positions.

The selection of content of thermite mixture and sizes of flare cartridge is required for dismantling of concrete equipment. The rigging by remote-controlled technics is required for cutting of radioactive equipment. At the present time the pilot installation for cutting of pipe-line have been created and tested.

- ***The method thermo-gas-jet cutting***

The given method is based on use of method of melting of material to be distracted by means of thermal and power action of high temperature supersonic jet, arising during combustion of hydrocarbon fuel (diesel fuel, kerosene, benzene) in oxygen media, and reactive oxygen burner (ROB), which is the generator of mentioned high temperature jet.

The robe is similar to the camera of burning of liquid reactive motor on principle of action and working principle (Fig. 3).

At the present time the demonstration installation has been developed and tested which can be used as independently as in combination with other technologies. The installation is easily rigged by manipulation devices.

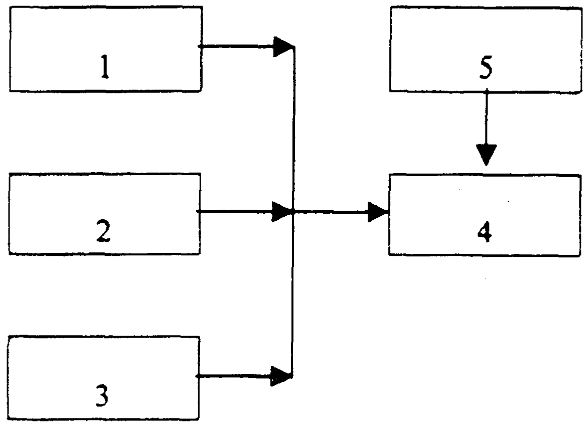


Fig. 3. Principal schematic diagram of thermo-gas-jet cutting:
1 - fuel supply system, 2 - oxygen supply system,
3 -cooling system, 4 -cutter (executor),
5 - cutter holder bar

- ***The technology of cutting by explosion***

The technology of cutting by explosion (TCE) is one of the high productivity technologies which can be used for dismantling of equipment of nuclear installation and which allow to reduce irradiation of personnel. The TCE can be designated for the cutting and chopping of different metallic and also concrete and ferro-concrete objects of different form. The TCE is based on use of industrial lengthened cumulative blasting charges of different kinds: string, linear, flexible in polymer coat, solid blasting charges in copper, steel and aluminium coats. The thickness of metal to be cut in the case of use of TCE can be from 5 to 60 mm.

- ***The technology of electro-erosion fragmentation***

The electro-erosion technology fragmentation, for example, of technological channel of reactors of type RBMK is based on combination of simultaneous use of method CAC, applied for separation of non radioactive parts from radioactive ones, and technology of electro-erosion (electro-spark) destruction of radioactive parts up to insoluble radioactive slime (in form of metallic powder) (Fig.4)

Later on this radioactive metallic slime can be gathered, compacted and buried. The coefficient of compactisation of solid radioactive waste can reach 0,6-0,7, moreover the accompanied secondary solid and liquid radioactive waste are practically totally absent in this technological process.

- ***The technology of laser cutting***

This technologies are based on the principle of controlled thermal action of high energy impulses of laser emission with specially selected parameters, resulted in evaporation and melting of metal in zone of cutting with possibility of next removal from zone of cutter and use of melt of metal and oxide films containing radionuclides.

At the present time the laboratory stand have been created on which the use of technology of laser cutting of metal-construction have been tested. The development are carried out on creation of multifunctional laser complex for remote cutting of pipe-lines and metal-construction, their deactivation during decommissioning works.

The impulse laser module on the base of crystals of allumo-yttrium garnet activated by neodymium with average efficiency of emission not less than 250 Wt and frequency of sequence of impulses from 50 to 250 hertz is supposed to use as a source of laser emission of projecting complex.

The preliminary estimation of possibility of use of laser technologies during decommissioning of nuclear installations show that use of this technologies is more effective for elimination of facing coats made from stainless steels, from hermetic compartments of units after carrying out of dismantling of equipment located in it.

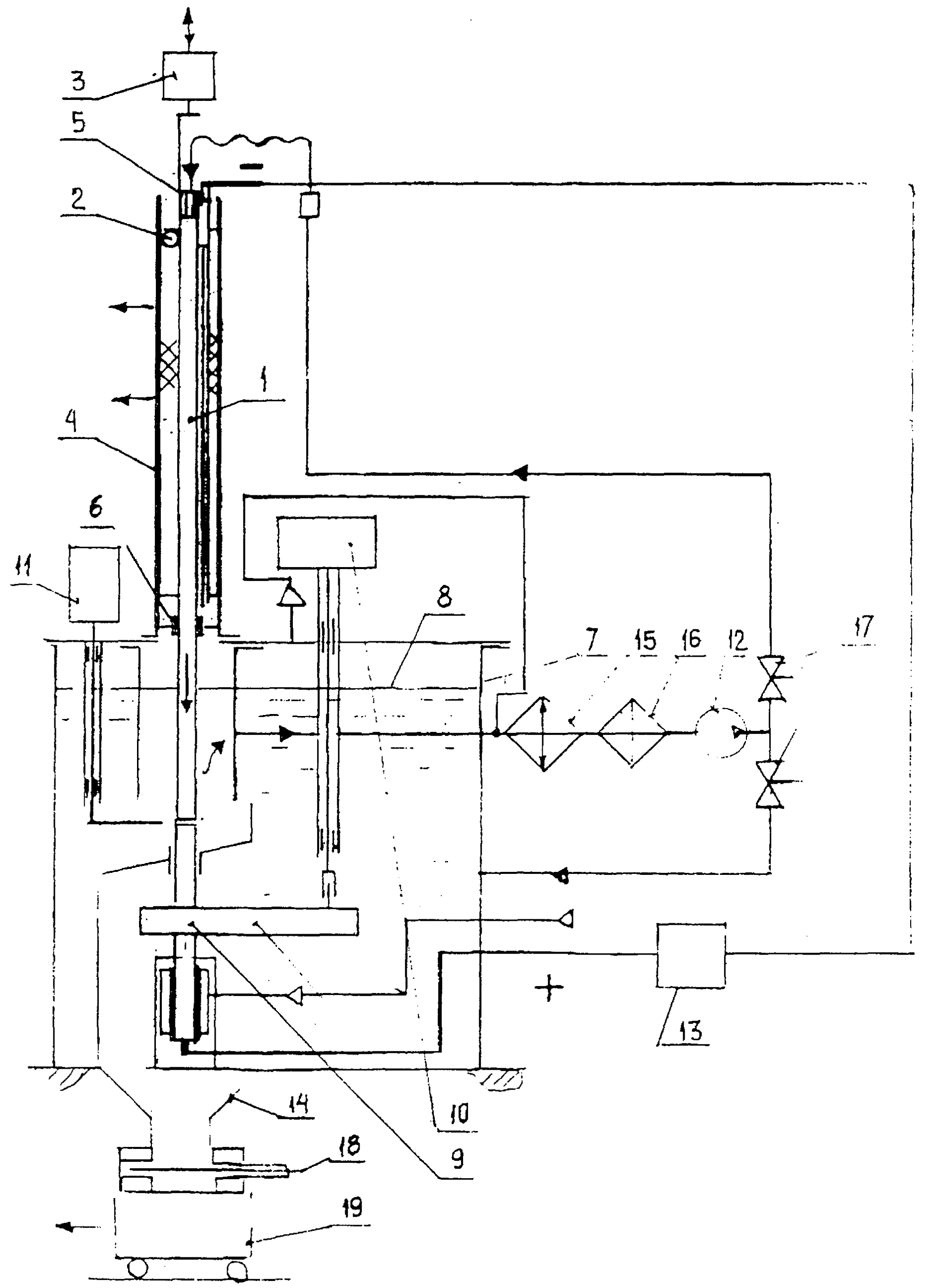


Fig. 4. The principal schematic diagram of TEF
1-channel (top fragment), 2-capture of vertical movement drive, 3- vertical movement drive: 4- finishing shaft, 5-plug, 6-seal, 7- finishing tank, 8-water level, 9- capture of rotation drive, 10-rotation drive, 11-electrocontact cutter, 12-pump, 13-electric generator, 14-precipitation tank, 15-separater-heat exchanger, 16-filters (mechanical and ion-exchange), 17-control valves, 18-gate, 19-transport container

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EUNDETRAF

Chapter 9

Decontamination techniques

Chapter summary

Decontamination techniques

9.1. Introduction

The contamination of materials and components is a process which covers various actual physical and chimico-physical aspects. Contamination can concern the deposit of some micrometer of contaminating product (i.e. a layer containing the radio-emitting nuclides) in the case or metals, up to a few centimeters in the case of concrete for instance.

The degree of attachment of the contaminant to the surface depends on a lot of factors, like the type of the base material, the surface roughness, the degree of corrosion, the surface or material porosity, as well as the physico-chemical properties of the fluid which was in contact with the contaminated material, like the pressure, temperature, pH, etc...

Depending on all of these factors, the decontamination process can be different or can have different results. Moreover, the objective of the decontamination can also influence the selection of the process to be used. This objective can be a reduction of the radiation dose rate of the component, the removal of loose or semi-loose contamination before dismantling or opening of a component to avoid a further risk of spread during subsequent dismantling, or the decategorization of the waste produced, or even the free release of the treated material, if the free release is an accepted and regulated process in the country where it is applied. Decontamination can also be used for maintenance purposes.

The decision to decontaminate should be weighted against the total dose and cost of the operation itself. Indeed, the decontamination operation will generate secondary waste (where the contamination has been concentrated), it involves costs for the operation and can also lead to radiation or internal contamination exposure, which must be balanced versus the potential or expected results and advantages of the operation.

9.2. Definitions and objectives

Decontamination is defined as the removal of contamination from surfaces of facilities or equipment by washing, heating, chemical or electrochemical action, mechanical cleaning, or other techniques. In decommissioning programmes, the objectives of decontamination are:
- to reduce radiation exposure;
- to salvage equipment and materials;
- to reduce the volume of equipment and materials requiring storage and disposal in licensed disposal facilities;
- to restore the site and facility, or parts thereof, to an unconditional-use condition;
- to remove loose radioactive contaminants and fix the remaining contamination in place in preparation for protective storage or permanent disposal work activities; and
- to reduce the magnitude of the residual radioactive source in a protective storage mode for public health and safety reasons, to reduce the protective storage period or to minimize long-term monitoring and surveillance requirements.

Some form of decontamination is required in any decommissioning programme, regardless of the form of the end product. As a minimum, the floor, walls, and external structural surfaces within work areas should be cleaned of loose contamination, and a simple water rinsing of contaminated systems may be performed. The question will arise, however, whether to decontaminate piping systems, tanks and components.

A decontamination programme may also require a facility capable of treating secondary waste from decontamination (*e.g.,* processing chemical solutions, aerosols, debris, etc.) The concentrated waste, representing a more significant radiation source, must be solidified and shipped for disposal in licensed disposal facilities unless properly treated in the waste reduction/recycling/ reclamation processing alternative. The optimal waste reduction configuration must be defined after an economic assessment of treatment *versus* transportation/disposal costs has been completed. Each of these additional activities may increase:

- occupational exposure rates;
- the potential for a release;
- the uptake of radioactive material.

These could conceivably result in even higher doses than those received from removing, packaging and shipping the contaminated system without extensive decontamination. Resolution of this question depends on specific facts, such as the exposure rate of the gamma-emitting contamination, the contamination level, and the effectiveness of the containing component and piping (wall thickness) in reducing radiation fields in the work area.

9.3. Selection of a decontamination process

Several decontamination techniques have been developed to support maintenance work in nuclear installations, and have been applied with relative success to decommissioning. Nevertheless, with the decommissioning operations becoming more and more carried out throughout the world, specific processes have also been developed for decommissioning purpose. The different type of decontamination processes are summarized in table 9.1. [from ref. 9.2]. for the most common aspects of decommissioning.

Table 9.1. – Decontamination for decommissioning

Decontamination before dismantling - Reduction of occupational exposure	➢ Pipe line system decontamination ➢ Pool, tank	➢ Chemical method ➢ Mechanical method ➢ Hydro Jet method ➢ Blast method ➢ Strippable coating method etc.
Decontamination after dismantling – Recycle of contaminated metal – Reduction of radioactive waste	➢ Pipes, Components	➢ Electropolishing method ➢ Chemical immersion method ➢ Blast method ➢ Ultrasonic wave method ➢ Gel method
Decontamination of building –Unconditional release of building –Reduction of radioactive concrete waste	➢ Concrete surface	➢ Mechanical method · Scabbler · Shaver · Drill & Spawling · Steel grit blast ➢ Thermal stress method · Microwave irradiation · Flame scarfing

Presently, the interest of the nuclear industry is moving from decontamination techniques for maintenance to decontamination for decommissioning. Limited data are available from decommissioning on the efficiency of usable techniques to meet the low unconditional-release criteria. In most cases, using available techniques, the clearance levels are only met in an asymptotic way. Not all methods and techniques available present the possibility of decontaminating to below the required clearance levels. So, in some cases, decontamination is carried out in different stages, the last step specifically aiming to obtain the required objectives.

Based on these considerations, when selecting a specific technique for system and/or component decontamination, mainly the following requirements must be considered [9.2.]:

- **Safety** – The application of the method should not result in increased radiation hazards due to external contamination of workers or even inhalation of radioactive dust and aerosols formed during its implementation; it should not add other hazards (*e.g.,* chemical, electric, etc.).
- **Efficiency** – The method should be capable of removing radioactivity from a surface to the level which would enable hands-on work instead of robotics, or which would permit recycle/reuse of material or, at least, a lower waste treatment and disposal category.
- **Cost-effectiveness** – Where possible, equipment should be decontaminated and repaired for reuse; however, the method should not give rise to costs which would exceed the costs for waste treatment and disposal of the material, whether including replacement of the equipment or not.
- **Waste minimisation** – The method should not give rise to large quantities of secondary waste, the treatment and disposal of which would result in excessive requirements for manpower and costs, thereby causing additional exposures.
- **Feasibility of industrialisation** – Due to the large quantities of contaminated materials involved, methods or techniques should not be labour-intensive, difficult to handle, or difficult to automate.

Other aspects are influencing the selection of the decontamination process, and we should advise the reader to refer to [9.2.] to look for details in the selection process and related topics.

9.4. Decontamination of metallic pieces

The decontamination of metallic pieces and equipments concerns mostly pieces made in stainless steel (SS) or in carbon steel (CS). Other materials like Aluminium (Al) or even less used metals like Inconel, Copper, lead, or other alloys are a bit less generic and require specific attention for the selection of the necessary process.

In [9.2.], a list of processes has been identified as being of interest for decontamination (Table 9.2). For decontamination of metals, the processes are divided into chemical, electrochemical and physical processes. Moreover, a distinction has been made between the processes used in closed systems (*e.g.,* full-system decontamination of the primary circuit of a reactor or the partial decontamination of closed loops), and the processes used in open tanks (*e.g.,* decontamination of dismantled pieces).

Table 9.2. – Overview of decontamination processes for decommissioning

Metal decontamination	Closed systems	Open systems	Metal decontamination	Closed Systems	Open systems
Chemical processes			***Physical processes***		
Oxidation processes			Ultrasonic cleaning		X
ODP/SODP	X		High pressure water		X
Cerium/Sulfuric acid		X	CO_2 ice blasting		X
Cerium/Nitric acid		X	Ice water		X
Oxidation-reduction proc.			Freon substitutes		X
APCE/NPOX	X	X	Abrasives wet	X	X
TURCO	X	X	Abrasives dry		X
CORD	X	X	Grinding/planing		X
CANDEREM/ CANDECON		X			
CONAP		X	***Combined mechanical/ Chemical processes***		
AP/NP+LOMI for PWR	X		Pastes + HP cleaning		X
EMMA	X		Foams/Gels/HP cleaning		X
Lomi for BWR	X		Vacuum cleaning (dry/wet)		X
Phosphoric-acid-based Processes		X			
Foams	X				
Various reagents					
HNO_3		X			
HNO_3 + HF	X	X			
$HNO_{3/NaF}$	X	X			
HCl	X	X			
DECOHA		X			
Electrochemical proc.					
Phosphoric acid		X			
Nitric acid		X			
Nitric acid-Electrodeplating		X			
Sodium sulphate ELDECON proc.		X			
Oxalic acid		X			
Citric acid		X			
Sulfuric acid		X			
Other electrolytes		X			

X = decontamination technique applied for open or closed systems

9.4.1. Full System and closed system Decontamination

9.4.1.1. Objectives

The most important objectives of a full or closed system decontamination are:
- to reduce the dose rate around the concerned loop
- to minimize the radiation dose related to the operation
- to concentrate the removed activity in as less as possible secondary waste
- to use the installation and its loops by carrying out a minimum of modifications.

9.4.1.2. Some of the most used processes

Following [9.3. and 9.1.], the most used or known processes for full system decontamination are summarized in table 9.3. with their principal characteristics.

Table 9.3. Processes for closed systems decontamination

Process name	General Description
LOMI	LOMI is an acronym for Low Oxidation State Metal Ion and was developed by the scientists at the Central Electricity Board (CEGB) in England in the late 1970s and early 1980s. The process incorporates vanadium (II) as a reducing agent and picolonic acid as the complexing or chelating agent. LOMI has been the most successful process for the removal of deposits where zinc and hydrogen water chemistries (HWC) have been employed during reactor operations. It is also the only process approved by the Electrical Power Research Institute (EPRI) and General Electric (GE) for use on GE designed reactor systems including the reactor pressure vessel and fuel.
LOMI-2	Similar to the properties outlined above for LOMI, but adjusted to be applied in a regenerative mode. The process, developed by EPRI in the late 1990s, reduces the secondary waste produced from the decontamination.
CANDEREM	A regenerative process comprised of citric acid and EDTA was developed by the Atomic Energy of Canada (AECL) in the mid-1980s. The CANDEREM process was used for the full system chemical decontamination performed by PN services at Indian Point 2 in the mid-1990s. The process is now approved by Westinghouse for full system decontaminations, with fuel in place for Westinghouse PWRs.
CITROX	A dilute regenerative process to be applied to both PWR and BWR reactor piping and system components developed in the 1980s. The CITROX process comprises citric acid and oxalid acid.
NITROX	A proprietary chemistry of PN services was developed in the mid-1990s for the chemical decontamination of reactor coolant pumps (RCPs). The cyclic process containing nitric acid, oxalic acid and ptassium permanganate was modelled on the CITROX process and was developed to minimise secondary waste. The NITROX process was qualified by Westinghouse specifically for the chemical decontamination of Westinghouse RCPs.

NITROX-E	Similar to the properties outlined above for the NITROX process but adjusted to destroy the chelating species during the process. The NITROX-E chemistry has been applied very successfully to both reactor coolant pumps and contaminated systems since its inception in the late 1990s.
REMCON	Is a family of chemistries employed by PN services for very specific customer applications.
AP and NP	Alkaline permanganate s(AP) and nitric acid permanganate (NP) are oxidation processes applied when radioactive deposits contain high levels of chromium. These processes were developed in the early 1980s and are used when the presence of chromium in the deposit renders the deposit insoluble by simple acidic dissoloution. Remnant testing prior to the chemical decontamination, or samples taken during the process, can determine when these chemistries need to be applied.
DfD	Or Decontamination for Decommissioning was developed by EPRI primarily for the decontamination of reactor systems and components for free release. The process was developed in the late 1990s and used by PN services for the full system decontamination for decommissioning at Big Rock Point in Michigan and at Maine Yankee.
CORD	The CORD process (Chemical Oxidizing Reducing Decontamination) developed by Siemens KWU is a three steps chemical process. It is applied in several cycles. Each cycle comprised the following steps: Oxidation step, using permanganic acid; Decontamination step using oxalic acid; Purification step by addition of permanganic acid or hydrogen peroxide.
EMMAC	Decontamination operations for following SG replacement were performed by Framatome-ANP in association with its affiliated company STMI using a dilute chemical process developed and patented by EdF (French Utility), called EMMAC. This same process is also used for the decontamination of chemical and volume control system heat exchangers. Today this process is used at the SOMANU (Framatome-ANP subsidiary) hot repair and maintenance workshop at Maubeuge in France. This facility consisting of four 1m³ tanks and two parallelepiped tanks, 4.2 metres long, is used to decontaminate equipment and components for disassembly/assembly and machining operations. Approximately 10 primary cooling pumps per year which arrive for repairs and with an initial activity level of 100 GBq are decontaminated by up to a factor of 50 using the EMMAC process.

One can distinguish two large types of processes: the so-called "hard" processes with a strong concentration of reagents, and the "soft" processes with a low reagents concentration. The hard processes are practically no more used because they lead to a higher secondary waste volume production. A few are only cited: TURCO, MOPAC and CITROX.
To give an idea of the details of a typical application, one will use a typical actual case of a full system decontamination, applied on a reactor before dismantling: the CORD process applied at the BR3-PWR reactor.

Description of the CORD process, used at BR3 for the decontamination in 1991

The CORD process (Chemical Oxidizing Reducing Decontamination) developed by Siemens KWU is a three steps chemical process. It is applied in several cycles. At BR3, three decontamination cycles were carried out.

Each cycle comprised the following steps:

- **Oxidation step:** For the crud present in PWRs, this step is necessary to oxidize the insoluble chromium. The permanganic acid $HMnO_4$ was used at a concentration of 0.4 g/l in order to oxidize the Cr^{3+} and Cr^{6+}.
- **Decontamination step:** After oxidation, a reduction step is carried out using oxalic acid at a concentration of 3 g/l. The oxalic acid first reduces the permanganic acid and then dissolves the crud layer. The iron, nickel, chromium and manganese as well as the activity are deposited in anionic or cationic form in the solution. The dissolved ions are fixed on the ion exchange columns placed in series.
  The activity and the cations are fixed on the cationic columns and the anions (complex oxalates) on the anionic columns.
- **Purification step:** Before starting a new decontamination cycle, the solution containing an excess of oxalic acid is treated, whether by adding permanganic acid, or by adding hydrogen peroxide. In this way, the oxalic acid is destroyed and transformed in CO_2 and H_2O. For the first two decontamination cycles, the destruction with permanganic acid was used and, for the last cycle, the destruction with hydrogen peroxide was used. For the two first cycles, the solution has to pass again on the cationic resins to eliminate the Mn^{2+} that has been formed. For the last cycle, the solution was sent to a column containing at the same time cationic resins and special anionic resins loaded with an oxidation catalyser.

Before carrying out the decontamination operation itself, some other operations have to be done: closing of the primary loop, control and maintenance of the primary and associated loops, and modifications on the loops. The most important modifications were the following:

- As the decontamination operation was carried out at low temperature (80 to 100°C), the working pressure in the primary loop was reduced to 20 bars.
- The thermal losses of the primary pumps were used to heat the solutions and the cooling was assured by the steam generator, of which the steam side was cooled by the tertiary cooling loop.
- The Siemens mobile decontamination station AMDA was connected to the purification loop, supplementary ion exchange columns were installed and some modifications were brought to the loop in order to connect the three columns available at BR3.

For the BR3 system, 3 successive decontamination cycles at an operating temperature of about 80°C to 100°C were performed. The operation itself lasted nine days of continuous operation. The following main systems were decontaminated : the reactor primary loop (fuel unloaded but internals loaded), the purification loop and a part of the Residual Heat Removal System (the Shutdown Cooling Heat Exchanger and the Emergency Shutdown Condenser).
The total volume of the loop is about 15 m^3 and the total surface treated about 1200 m^2; most of the surface, steam generator included, is of Stainless Steel 304. During the 3 cycles, a total of 2 TBq of gamma emitters was released with ^{60}Co as the dominating γ nuclide, which is no less than 90% of the activity of the primary circuit (Ref.date 1991). A total quantity of about 2.3 GBq alpha activity was also removed from the circuits among which about 185 mg (equivalent to 629 MBq) of plutonium. A

total quantity of about 33 kg of oxides or corrosion products was removed; this corresponds to a mean release of about 2.8 mg of oxide/cm^2. A mean Decontamination Factor close to **10** has been achieved, with a broad spread of individual values ranging from 0.1 (redeposition of activity in a horizontal pipe) to 31 (steam generator) according to the measurement location.
In the plant container, the decontamination operation has deeply modified the picture, as far as working conditions are concerned: the ambient dose rate has been reduced by a factor about 10 and varied after the operation between 20 and 60 μSv/h. In the purification circuit, the ambient dose rate was then around 10 to 30 μSv/h. [9.5.]

For the CORD process, it has to be noted that this process has evolved since its application at BR3. The chemistry of the process remained the same but the process was improved in order to reduce the produced secondary waste volume.
The modified CORD process to be used as decontamination process in view of dismantling presents the following steps:
- oxidation step with permanganic acid
- reduction step with oxalic acid with an increased concentration of 10 to 20 g/l
- fixation of the activity and the cations on the cationic exchange resins
- purification of the solution by destruction of the oxalic acid with hydrogen peroxide in the presence of UV in a special reactor
- fixation of the Fe, Cr cations on cationic resins.

At the end of a cycle, the Fe and Cr can be eluted in a selective way, as the elution solution can be treated by evaporation in the presence of hydrogen peroxide.
The process can be used either in a loop to be dismantled, or on dismantled pieces.

9.4.2. Decontamination of subsystems, equipments, pieces after dismantling

The aim of these processes is to allow a decategorization (i.e. change of the waste category towards a lower category or class, which is cheaper to dispose of) of the waste and even a potential unconditional free release, in order to allow a drastic reduction of the total produced metallic waste volume.
Most of these processes are used in discontinuous process, working either in bath with *immersion of individual pieces in the baths or in batch.*

One can refer to table 9.2 above, column "Open systems" to have an overview of the most common processes used nowadays for this purpose.
Some of them will be analysed more in detail in the subsequent paragraphs. They are classified following their working principle (i.e. chemical, electro-chemical, mechanical, etc...).

9.4.2.1. Chemical processes

The processes are briefly described.

9.4.2.1.1.Processes in several steps

The layer of corrosion products forming the typical crud of PWR reactors, is very insoluble and is characterized by a strong content of chromium oxides. These processes always use an oxidation step followed by a dissolving step of the oxides and a *complexation* of the dissolved metals. Many processes exist but the nature of the used reagents and/or the concentration of the reagents differ.
Like for closed systems decontamination, one can distinguish two large types of processes: the so-called "hard" processes with a strong concentration of reagents, and the "soft" processes with a low reagents concentration. The hard processes are practically no more used because they lead to a higher secondary waste volume production. The soft processes are used for the decontamination of loops, entirely or partially. The most frequently used processes are given in table 9.4. with their most important characteristics.

Table 9.4. – Most frequently used processes

	CANDECOM CANDEREM	LOMI	CORD
Oxydation	AP NP 95°C	AP NP 90°C	$HMnO_4$ 100°C
Reduction	$H_2C_2O_4$	$H_2C_2O_4$ HNO_3	$H_2C_2O_4$ 80-90°C
Decontamination	Oxalic acid Citric acid EDTA	LOMI reagent Vanadous picolinate formate	$H_2C_2O_4$
Purification	Ion exchange resins	Ion exchange resins	Ion exchange resins

AP = Alkaline permanganate
NP = Nitric acid potassium permanganate

These processes are mainly used for the decontamination aiming at a dose rate reduction. In general they are carried out in several steps. The CORD process, used for the decontamination of the primary loop was also tested for the decontamination of dismantled pieces.
When these processes are used in a bath on cut pieces, their efficiency can be improved by a combination of chemistry with ultrasounds.

In a general way, these processes are especially well adapted for decontamination in the framework of reactor operation, in view of reusing the equipments. As a result of their weak aggressivity, they do not have an influence on the integrity of complex systems (no problems with the gaskets, seals, valves or critical parts).
On the contrary however, for the decontamination of cut pieces in a bath with a view to their dismantling and thus their possible release as non radioactive waste, it is recommended to use more aggressive methods attacking also the base metal, as this gives a better assurance regarding the final level of contamination reached. A supplementary drawback of these techniques in several steps is the fact that, in general, they demand the application of several decontamination cycles and, consecutively, more decontamination baths placed in series. The pieces are passing from one bath to another, then a new cycle is restarted until the expected residual levels are reached.

9.4.2.1.2. Processes in one single step

Here a synthesis will be made of the chemical processes applied in one single step and using sufficiently aggressive reagents in order to reach the residual levels allowing the free release of the treated pieces.

Process with Cerium4+
The Cerium4+ process uses the potential of raised oxidation of the Cerium4+ (+1.61 V/ENH) to assure at the same time the oxidation of the oxides present in the crud layer (chromium of valence 3 to valence 6 and the oxide of ferrous iron into ferric iron) as well as to oxidize and to put in solution a base metal layer of some microns. This double attack guarantees the complete decontamination of the piece as far as the reagents can reach all contaminated surfaces. The Cerium4+ process was mainly developed at Studsvik in Sweden, they defined the SODP process (Strong Ozone Decontamination Process) and at JAERI and JPDR in Japan, they developed the REDOX process, and at SCK•CEN in Belgium, where the MEDOC process was developed.

The SODP process
This is a one step decontamination process in view of dismantling PWR type reactors. The process works at ambient temperature and is based on the use of a nitric acid environment with a pH of approximately 0.6. The oxidant used is Cerium+4 regenerated by addition of ozone. The Cerium reduced to valency 3 is reoxidized by ozone at valency 4.
This process was used in Sweden to decontaminate heat exchangers and steam generators and was tested in France on a steam generator of Dampierre.
The treatment of the solutions after decontamination consists of a reduction of the overrun Ce^{4+} to Ce^{3+} by using hydrogen peroxide followed by a precipitation of the hydroxides in a basic environment.

Due to the working temperature of 20°C, this process is relatively slow. It is well adapted for the decontamination of dismantled steam generators from power reactors, for which the application of processes with higher temperatures may cause some problems.

The REDOX and SC (Sulfuric Acid Cerium) processes

These two processes, developed in Japan, are based at the same chemistry, namely the use of the oxidizing power of Cerium4 and a decontamination temperature of 60 to 80°C, thus accelerating the reaction compared to an application at ambient temperature as the SODP process. These two processes use an electrochemical regeneration of Cerium3 reduced by the reaction. The electrochemical reactor is separated from the decontamination reactor, the decontamination solution is continuously regenerated. These processes have mainly been used for the in bath decontamination of complex pieces.

After decontamination, the pieces are taken out of the decontamination bath and rinsed in an ultrasound bath. The decontamination solution is used till the concentration of dissolved salts exceeds approximately 10 kg/m^3. Indeed, the dissolution of salts reduces at one hand the electrochemical performance of the regeneration and, on the other hand, it has to stay beneath the solubility limit of the dissolved salts. Moreover, also the activity of the solution increases.

The MEDOC process

The SCK•CEN has improved existing processes and has developed a process called MEDOC (MEtal Decontamination by Oxidation with Cerium).

This process distinguishes oneself essentially by the continuous regeneration of the solution at the same temperature as the decontamination temperature, and it is realized with ozone in a gas-fluid contactor.

This process combines in fact the advantages of the two existing processes: an accelerated attack rate by working at an increased temperature and a simple regeneration technique using ozone.

The secondary waste volume is low. It is estimated at 11.5 l of bituminized waste per ton of treated material for 20 m^2/ton and 10 μm dissolved.

This volume could be further reduced by recycling the sulfuric acid. It can be done for instance by electro-dialysis of the solutions.

The regeneration by electrochemical means was not chosen because it presents some disadvantages compared to the ozone process:

- hydrogen production
- problems regarding the maintenance of the electrolyzer module
- long-term resistance of the electrodes.

This process is used for the in bath decontamination of stainless steel pieces.

The installation comprises mainly:

- a decontamination tank with a basket filled with pieces to be decontaminated
- a buffer tank and a gas-fluid contactor for the ozone regeneration; the ozone is produced by an ozonizer supplied with oxygen
- a rinsing tank with ultrasounds.

The results give Decontamination Factors larger than 10000.

The HNO_3/HF process

The sulfonitric mixture is commonly used for the etching of stainless steel. In this case, it is applied either in a bath, by pulverization using a pressure jet, or by application of an etching paste.
In the case of stainless steel covered with an oxide layer, the attack mechanism is a combination of an oxides reducing attack followed by the dissolution of the metal underneath, thanks to the penetration of the liquid through the oxide layer.
By the attack of the base metal, the oxides come off and stay in the solution. The solution is thus progressively loaded with insoluble oxides in the form of particles, and with dissolved salts coming from the partial dissolution of the oxides and the transformation of the base metal into solution.

In bath process or solutions pulverization

This process has been tested for the thorough decontamination of dismantled metallic pieces. The following table 9.5. gives the typical operational conditions mentioned in the literature.

Table 9.5. - Fluoronitric processes

Process parameters	**ENEL Italy**	**CEA France**	**BelgoProces Belgium**	**SCK•CEN Belgium**
Temperature	40	25	40	
HF M	0,5 to 0.75	0,5	2.5	
HNO_3 M	0.4 to 0.8	2	4.5	
Duration h	4 to 6	2 to 8	1	
Temperature	80	85		60 - 95
HF M	0.15 to 0.25	0.05		0.25 - 0.5
HNO_3 M	0.4 to 0.8	2		0.5
Duration h	5	1		1 to 3

After treatment, the piece is rinsed with a pressure jet or in a rinsing bath with ultrasounds.

The speed and the efficiency of the reaction increase with the temperature, the concentration of HF and the reaction time. The process is generally used at low temperature and with a strong concentration in pulverization on large pieces (i.e. inner surface of tanks etc.), and at higher temperature and lower concentration on pieces put in a bath.
This process allows the complete decontamination of pieces even if strongly contaminated. The removal of an oxide layer and of a metal thickness of 10 to 20 μm allows indeed to remove the contamination completely. The specific activity reduced from a value of 100-5000 Bq/g to less than 0.3 Bq/g of ^{60}Co. The global efficiency of the process is improved when, after treatment, the piece is rinsed in an ultrasound bath.
Part of the oxides remains insoluble and can be eliminated by filtration.

The efficiency of the treatment decreases progressively with the increasing concentration of dissolved salts in the solution. The aggressivity of the fluoric ions decreases by chelation of the fluoric ions to the dissolved metal. Therefore, new HF has to be added to the solution or the bath has to be renewed.

The treatment of the fluoronitric solutions is generally a neutralizing and chelating treatment of the remaining free fluorides. The following table 9.6. gives the applied treatment methods.

Table 9.6.: Treatment techniques of the fluoronitric solutions

ENEL Italy	CEA France	BP Belgium	SCK•CEN
Neutralization NaOH Precipitation of the hydroxides	Neutralization $NaOH$, $Ca(OH)_2$ $CaCL_2$, ferrocyanide Chelation of F Precipitation of the hydroxides	Chelation of F by $Al(NO_3)_3$ Neutralization by $NaOH$, $Ca(OH)_2$ Precipitation of the hydroxides	Same method as Belgoprocess Solution sent to BP for conditioning
Cementation of filters and sludge	Cementation of filters and sludge	Incorporation in bitumen	

The in bath technique is not very advantageous. It needs the use of special construction materials (mainly plastic materials) and it presents important problems regarding the safety of the workers. Moreover, the efficiency of the bath decreases rather quickly and leads to an important consumption of reagents.
On the contrary, the pulverization of a film on the walls of an equipment to be decontaminated can be very advantageous. The quantity of used reagents is very limited, the application at low temperature gives less security problems and the volume of produced effluents is smaller.

Process with etching paste
The same process can be used by applying an etching paste on the surface of the object to be treated. After a few hours, the paste is dry and can be removed mechanically. The treatment finishes by rinsing with a pressure water jet.
This technique is mainly used for external surfaces or for places of local contamination. It has been tested with success on laboratory scale and could be applied, for instance, for the decontamination of pieces coming from dismantled equipment. This technique will allow to get the equipment out, or to carry out an *etching-passivation after it has been used for a long time in the pool water.*

The DECOHA process
This process is based on the use of fluoroboric acid.
The metals react as follows:
$n\ HBF_4 + Me \Rightarrow Me(BF_4)n + n/2\ H_2$

The oxides react as follows:
$n\ HBF_4 + \text{metal oxide} \Rightarrow Me(BF_4)n + H_2O$

This process is applied by pulverization of a solution at low temperature, or by immersion in a bath at a higher temperature, till approximately 90°C.
The used solution can electrochemically be regenerated. The cations are then reduced at the cathode and are deposited as metal.
$Me(n+) + n\ e \Rightarrow Me(0)$
At the anode, HBF_4 acid is reformed. The ^{60}Co co-precipitates with the metals at the anode.

The regeneration does not allow the 100% elimination of the cations, it is generally stopped when 98% of the cations are deposited. ^{137}Cs is not deposited at the cathode and stays in the solution. In order to treat the remaining 2% several techniques are possible:

- precipitation of insoluble phosphates by addition of phosphoric acid
- precipitation of insoluble oxalates or insoluble silicates
- neutralization of the solutions with $Ca(OH)_2$ giving rise to the formation of insoluble CAF2.

The concept of the DECOHA process is illustrated by applying it at industrial scale (5t/day) for the treatment of contaminated metallic pieces from Tchernobyl.
The steps of the process are the following:

- Metallic pieces are placed in the tank.
- The activity that is not fixed is removed by sprinkling with a volume of approximately 500 l of HBF4 solution at 10%.
- The tank is then filled with 15 m³ HBF4 at 20% at 90°C and the decontamination reaction starts; the reaction speed is about 0.2 to 1 μm/h. The reaction lasts from 3 to 18 h.
- The solution is drained and the pieces are rinsed with 500 l of cold water.
- The pieces are rinsed with 15 m³ of water; the rinsing water is regenerated by distillation.
- The regeneration is started when the concentration of dissolved metal reaches about 70 g/l. From this moment on, the work is done under stationary conditions, this means with a constant concentration of dissolved metals.

This decontamination process can be used for the decontamination of stainless steel (rate of 0.02 to 0.5 mg/cm².h), carbon steel (rate of 0.05 to 1000 mg/cm².h) and also for the decontamination of aluminium (rate of 6 mg/cm².h or 22 μm/h). Compared with other processes like the Cerium or the sulphonitric one, this process is less aggressive (lower rate). However, being less aggressive, it shows fewer security problems than the HF. The regeneration by cathodic precipitation allows to recycle the HBF4 and to deposit the major part of the cations at the cathode. The construction material recommended for this kind of product is essentially polypropylene. The liberation of hydrogen during the reaction requires the use of a ventilation, and dilution of the gaseous effluents in order to stay beneath the lowest explosion limit of the H_2-air mixture.

9.4.2.2. Electrochemical processes

The electrolytic polishing is an anodic dissolution technique. The material to decontaminate is the anode, the cathode being either an electrode in stainless steel or copper (helping electrode) in an adapted form, or the decontamination tank itself. During the decontamination, a controlled quantity of surface metal dissolves taking with it the contamination fixed in the surface layers. This technique is nowadays often used in fabrication, for the surface treatment of pieces in stainless steel in order to obtain a very good finish by decreasing the roughness. This improves the resistance to corrosion and allows also a better and easier future decontamination.
This process is also applied for the decontamination either in order to reduce the dose rate (typical case of the water boxes of steam generators), or for the decategorization of pieces to be dismantled.

If this technique is applied on pieces to be reused, the surface finish is an important parameter to reduce the possibility of a new contamination of the pieces. For pieces destined for the scrap, the surface finish is not so important as the efficiency of the decontamination and the volume of secondary waste produced.
The different used processes distinguish themselves by the used electrolyte and the operational conditions.

The phosphoric acid process

The representative parameters values for a decontamination using phosphoric acid are the following ones:

– H_3PO_4 concentration:	40 to 80% volume
– working temperature:	40 to 80°C
– potential difference:	8 to 12 V dc
– current density:	60 to 500 mA/cm²

This process can be applied to carbon steel and stainless steel.

An application at industrial scale has been conducted at the KRB plant of Gundremmingen (BWR of 250 MWe). Several hundreds of tons of ferritic steel have been completely decontaminated by using this process.
A thickness of 100 to 500 μm had to be removed in order to obtain a metal without any contamination.

The installation comprises two tanks of 3 m³ and a rectifier of 6000 A maximum. The current used was approx. 2000 A corresponding to a current density of 100 to 400 mA/cm² following the geometry of the pieces. The time of the treatment is about 2 h.
When the iron concentration in the solution exceeds 100 g/l, iron oxalate precipitates, carrying along a part of the ^{60}Co activity. In this way, one obtains a ^{60}Co enriched sludge compared to the solution.

The treatment of the phosphoric acid solution comprises the following steps:
- precipitation of the ferrous oxalate by addition of oxalic acid with entrainment of the ^{60}Co activity (^{137}Cs is not being precipitated);
- separation by sedimentation of the residual phosphoric acid, the diluted acid is re-concentrated by evaporation to be reused;
- the iron oxalate is destroyed by pyrolysis at 250°C in iron oxide that can be conditioned for final storage.

The treatment of 1 t (5 m²/t) of metallic waste produces 32 kg of iron oxalate and finally, after pyrolysis, 15 kg of iron oxide.
This process is very efficient for ferritic steel but presents a certain complexity for the secondary effluents treatment, in order to avoid the presence of oxalates in the conditioned waste. From a technical point-of-view and in the case of painted carbon steel, one has to consider also a pretreatment to remove the paint layer. This pretreatment could be chemical (NaOH) or mechanical. The process releases also H_2, so the gases have to be strongly diluted to avoid explosion risks.

The nitric acid process

Process developed by AEA and Harwell

This process was developed at Harwell and commercialized by AEA for the decontamination of stainless steel. Two systems were developed. On one hand, an immersion system for cut pieces and, on the other hand, a system with a small chamber for teleoperated decontamination. The latter one is particularly indicated for decontaminating hot spots in a high ambient dose rate and will not be discussed here.

The recommended operation conditions for decontamination are the following ones:
- the use of HN03, 1M at an ambient temperature
- a low current density, from 2 to 3 mA/cm^2
- the use of Ti electrodes to minimize the formation of H_2 and NOx.

This process has a high penetrating capacity; the decontamination is still effective in a closed tube at a depth of 5 times the diameter. The distribution of the current is easy: the pieces are put in a titanium basket. In 2 hours a DF of 10 000 is obtained, corresponding with putting a thickness of 5 μm in the solution. AEA announces a waste production of 0.6 dm^3 for each m^2 treated. The chosen environment, $HN0_3$, is compatible with the waste treatment installations of AEA.

Process developed by CEA/UDIN

The CEA is developing a anodic dissolution process in drum in an HNO_3 environment for the treatment of plutonium contaminated metallic waste. The waste in bulk is placed in a titanium perforated reactor, polarized in anode and put in motion. The stainless steel cathodes are put in parallel in the drum. Nitric acid is chosen for its compatibility with the treatment of the decontamination effluents and its reprocessing in case presence of PuO_2.
The experimental conditions for stainless steel pieces are the following ones:
- $HN0_3$ concentration of 2M
- current density of 10 to 30 mA/cm^2
- duration of the treatment: 1 to 3 h

The erosion rate in this conditions is about 16 μm/h and the alpha waste could be decategorized beneath the limit of 3.7 MBq/kg in order to authorize surface storage. This process is also applicable to unpainted ferritic steel, the erosion rate is then very high, about 500 μm/h.
It is also applicable aluminium alloy, the erosion rate amounts then to 20 μm/h.
The putting into service of this process of such a capacity or a larger capacity gives raise to some technological problems. This process is easily applicable for pieces of small dimensions presenting no "hidden" contaminated surfaces (internal sides of tubes, blind holes...).

Sulfuric acid process

This process is developed by Toshiba in Japan for the decontamination of stainless steel pieces in the framework of dismantling the Japanese reactors.
The characteristics of the process are the following ones:

- H_2SO_4 concentration of 5%
- temperature of 60°C
- current density of 300 to 1000 mA/cm².

At the beginning of the treatment, an erosion rate of 240 μm/h can be reached. When the dissolved salts concentration increases and the acidity decreases due to the consumption of H+, the erosion speed decreases to 60 μm/h.
The current efficiency reduces also with the concentration of dissolved ions, going from 40 to 20%. The acidity has to be kept at a certain level, sufficient to maintain the rate and the efficiency to an optimal value.
Exploratory tests have been carried out on strongly contaminated BR3 pieces. They showed that it is indeed possible to reach a high erosion rate till about 50 mg/cm². In order to obtain a very low residual contamination level, a quite large quantity of the material has to be removed. The contamination present in the crud does not dissolve completely and an important fraction remains as crud particles which were removed from the surface by the underneath erosion of the metal. To accelerate this phenomenon, it is better to change regularly the polarity of the electrodes.
Compared to the Cerium process, a thicker layer has to be removed in order to obtain the same residual contamination level.
The treatment of the solutions comprises the neutralization of the solution and the precipitation of the hydroxides. If ^{137}Cs is present, a treatment with ferrocyanide in combination with the precipitation of the hydroxides has to be applied.

<u>The sodium sulfate process</u>

The AEDSS process of Toshiba

Toshiba also developed a process based on the use of sodium sulfate as electrolyte and alternating the polarization of the electrodes. It is called AEDSS process, Alternative Electrolytic Decontamination with Sodium Sulfate. In anodic electrolysis, the metal is oxidized and dissolves, but oxides like ferric iron oxide remain intact. The attack of the metal situated underneath the oxide layer is only realized when the electrolyte penetrates through the film. With cathodic electrolysis, the oxide film at the surface is reduced, the ferric iron is reduced to ferrous iron and the base metal remains intact. A combination of the two techniques, this means a regular modification of the polarization of the electrodes, will accelerate significantly the global erosion rate.
This process is particularly interesting for pieces in ferritic steel covered by a thick layer of iron oxides.
The operational conditions are the following:
- Na_2SO_4 concentration of 20 wt%
- voltage 20 V.

The iron in the solution precipitates in the form of iron hydroxides carrying along the ^{60}Co.
The current efficiency is relatively weak of the order of 6% leading to a large energy consumption of about 4000 kWh/ton (200 kWh/m² for an erosion of 50 μm). On the contrary however, following Toshiba, the secondary waste is very low, about 10 kg/ton waste in the same conditions.

The ELDECON process from ABB Atom in Sweden

This process, based on the use of sodium sulfate as electrolyte, is commercialized by ABB Atom for the decontamination of reactor pieces.

The characteristics are the following:
- Na_2SO_4 concentration of 5 wt%
- current density from 100 to 600 mA/cm²
- pH = 7
- potential difference of max 24 V
- duration of the treatment: from 15' to one hour depending on the pieces, the erosion rate is approximately 60 μm/h at a current density of 400 mA/cm².

ABB is commercializing an installation with a bath of 1060 x 660 x 500 mm meant to accept a charge of 600 kg. The rectifier delivers maximum 1000 A at 24 V.
During the decontamination, the dissolved iron precipitates in hydroxide form carrying with it the ^{60}Co activity.

9.4.2.3. Physical processes

Besides the chemical and electrochemical processes used for thorough decontamination, the physical processes are often a bit simpler to use, but are generally less aggressive than their (electro-) chemical counterparts. The physical processes, as their name indicates, use a physical mean to remove the contamination layer from the base material.

Cleaning in ultrasound bath

The cleaning in an ultrasound bath is a classical technique for the decontamination of pieces presenting an unstable not strongly fixed contamination. It is generally used with addition of a detergent (like e.g. DECON90). The energy losses in the bath increase the bath temperature in a natural way, thus favoring the decontamination effect.
This technique is only applicable for slightly fixed contamination.
For pieces strongly contaminated like stainless steel pieces having stayed for a long time in storage pools, this technique does not allow to remove the contamination fixed on the piece. The DF are smaller than 2.

The cleaning in an ultrasound bath is very often used for the rinsing of pieces that underwent a chemical decontamination in order to eliminate completely the liquid film as well as the liquid micro droplets fixed at the surface of the material. This treatment is even more important as the roughness after chemical treatment is high.
Ultrasounds can also be used immediately in combination with a chemical treatment process. Generally in this case one observes a synergy between the effect of the ultrasounds and the chemical effect. This synergy increases the efficiency of the reaction and allows to shorten the treatment time. Moreover, the mechanical effect allows to loosen the crud particles and thus to liberate the attack surfaces.

Projection of CO_2 ice

This decontamination process uses the principle of projecting CO_2 ice pellets at high speed against the surface to decontaminate. The pellets are accelerated in an injector by means of compressed air at 18 bar and with a rate of about 21.5 m^3/min. The contamination is pulled out of the surface and carried away by the excess of air. The CO_2 pellets evaporate and the removed contamination is settled, or taken away by the air filtration system. Finally, the contamination will be located on the floor of the enclosure and/or in the filters.
The ventilation of the enclosure has to be sufficient in order to avoid accumulation of CO_2 and to eliminate air contamination. The operators are working in ventilated suits. The noise level can be from 75 dB to 125 dB.

Different results are reported, and it is strongly recommended to let make some decontamination tests before selecting this process for a particular operation.

Projection of ice

This technique is completely similar to the one with CO_2. Ice crystals are projected by means of compressed air against the surface to be treated. The water, formed by melting of the ice, takes the removed contamination away; the contamination is then in suspension or dissolved.
The putting into service of this technique is a little more easier than the CO_2 technique. The contamination is trapped in the water and requires a system to treat liquid effluents. The efficiency is comparable to the one of CO_2. This process is insufficiently aggressive to carry out a thorough decontamination of fixed contamination.

Pressurized water jet

Low pressure water jet – 50 to 150 bar

This technique is frequently used as pre-decontamination technique for pieces strongly contaminated. It is used for cleaning the pool walls, for pieces or tools having stayed a long time in the pool, and for the decontamination of dismantled pieces.
More particularly, this technique allowed to reduce the dose rate of dismantled activated pieces beneath the limit of 2 mSv/h by removing an unstable residual layer of crud. This technique is regularly used to decontaminate dismantling tools that are contaminated, for instance with activated metal swarfs or with crud deposition.

Medium pressure water jet – 150 to 700 bar

This technique allows the decontamination of surfaces presenting a cold fixed contamination. The water rate is from 60 to 6000 l/h depending on the apparatus. Additives can be added and can be used with warm water up to a temperature of 60°C. The efficiency increases with the working pressure. The water consumption can vary from 10 to 600 l/m^2. The work output for a large surface, for instance the pool walls, is approximately 10 to 20 m^2/h. This technique is usually used in nuclear plants for the decontamination of equipments in order to reduce the dose rate and the transferable contamination. In some cases, it allows thorough decontamination.

The drawbacks of this technique are the formation of contaminated aerosols and the large water consumption. This technique can however be considered if working with recirculation after filtration, or in an open loop in case it is used for specific decontamination operations like for instance removing contaminated equipment out of the dismantling pool.

High pressure water jet > 700 bar

This technique is identical to the one used for cutting metals. The water is pressurized at a pressure able to reach 3000 to 4000 bar. The water at the outlet of the injector reaches a speed of about 900 m/s. The efficiency can be increased by adding abrasives.
The water rate is about 120 to 240 l/h. The power of the compressor is about 20 to 40 kW. The investment for such a technique is rather high. Moreover, when using abrasives, the consumption of those is considerable as this system is not foreseen to recycle the abrasives.
This technique allows a thorough decontamination as it is possible to remove a layer of the surface material. The formation of liquid aerosols is relatively important as a result of the very high speed of the ejected water.

<u>Decontamination with abrasives</u>

This technique uses the power of the abrasives of different types projected at high speed against the surface to be treated.
Imperative for this technique is to assure the recycling of the abrasives in order to limit the secondary waste production. So the dry or wet sandblasting units without recycling are a priori excluded.
Two techniques seem more appropriate: decontamination by wet sandblasting with recycling of the abrasives, and decontamination by dry sand blasting with recycling of abrasives.

Abrasives in wet environment

In this case the fluid transporter is water and compressed air under pressure. The installation for a decontamination by wet sandblasting comprises:
- a ventilated working enclosure connected to the general ventilation system through HEPA filters
- a system to collect the mixture water-abrasives-contamination
- a sandblasting pistol with water jet/compressed air and addition of abrasives
- a recycling unit for abrasives with a circulation pump
- a filtration unit for the fine abrasives and removed contaminants
- a preconditioning unit for the filtration sludge.

This system is in use in various decommissioning projects and showed good results for large contaminated pieces, painted metals and localized (but reachable) contamination patches.

Dry abrasives

In this case the fluid transporter is compressed air.
An installation for dry sandblasting comprises:
- a working enclosure put under depression and filtration by pre-filters, type filter bags and connected to the general ventilation system through HEPA filters
- sandblasting pistols or injectors with compressed air
- a system to collect the abrasives and the contaminants
- a system to separate the abrasives and the contaminants; generally it is a cyclone system. The abrasives are collected in the cyclone and recycled. The contaminants and the fine abrasive particles are evacuated to the filtration system where they are collected and filtered.

Here also the system is in use in different decommissioning project with relative good results. One can mention e.g. the automatic system used at Belgoprocess (Belgium) where the metallic pieces are automatically loaded, decontaminated and unloaded, while the abrasives are recycled.

Comparison of the wet and dry sandblasting techniques

For these two techniques it is important to choose an abrasive with a long lifetime and a high decontamination power. The following abrasives have a decreasing hardness:
- shot of stainless steel
- ceramic micro balls or angular ceramic
- angular garnet
- glass micro balls.

The purchase price of the abrasive follows the same order. In spite of the higher price of the abrasive, it is important to choose an abrasive that deteriorates very slowly as this will reduce the secondary waste production.
More generally and depending on the application, a variety of materials may be used as abrasive media:
- minerals (*e.g.,* magnetite or sand);
- steel pellets, aluminum oxide;
- glass beads/glass frit, silicon carbide, ceramics;
- plastic pellets;
- natural products (*e.g.,* rice hulls or ground nut shells);

Silica has also been used as an abrasive; however, its use is not recommended since it is moderately toxic as a highly irritating dust and is the chief cause of pulmonary disease. Prolonged inhalation of dusts containing free silica may result in the development of a disabling pulmonary fibrosis known as silicosis.

The two techniques (wet and dry) allow to recycle the abrasive by separating the abrasive from the contamination. For the wet sandblasting, the contamination is trapped on the filters. For the dry sandblasting, the contamination is filtered on declogging filters and collected in a drum.

The two techniques have to be used in a ventilated enclosure. However, the air contamination is much more important using the dry sandblasting compared to the wet. The risk to contaminate the walls of the enclosure is greater, cross contamination can be provoked when changing from strongly contaminated pieces to slight contaminated pieces. But this system can be automated.

Decontamination by grinding, polishing, brushing

There are several devices on the market. Generally, the same device can be used for different functions and for different types of surfaces. A large range of abrasive belts or rollers are available. Due to the production of dust, this type of contamination has to be carried out in a ventilated enclosure, and the operator has to wear protection clothes (mask or ventilated suit).

9.4.2.4. Techniques based on the use of the melting of metals

The melting technique can also be considered as a decontamination technique. Indeed, certain elements are found in fumes and dust (i.e. ^{137}Cs), others in slag and are eliminated as waste (i.e. heavy weight elements that form oxides like uranium and plutonium). Elements like ^{60}Co or ^{63}Ni follow the metal in ingots.

The melting of metallic waste can be considered for:

- the recycling by melting of metallic materials with reuse of the recycled materials in the nuclear field, for instance containers for radioactive waste
- to allow by melting a precise radiological estimation of the residual activity of the decontaminated pieces and also to allow their clearance.

For information, table 9. 7. gives a summary of the most important melting facilities in Europe and in the U.S., with their principal characteristics [X.3.]. Two other are also existing, but few information could be gathered about them: the Capenhurst Melting Facility, United Kingdom (start 1994), and the Manufacturing Sciences Corporation (MSC) in Oak Ridge, USA (start 1996).

Table 9.7. – Characteristics of operating melting facilities

Facility	Furnace type	Types of metal treated	Charge size	Products	Radiological limitations
INFANTE	Electric arc melting furnace	Carbon steel, stainless steel	12 t	Ingots, shield blocs, waste containers	Max. 250 Bq/g for ^{60}Co, other limits for other nuclides
STUDSVIK	Induction for steel, small electric arc for aluminium	Carbon steel, stainless steel, aluminium	3 t	Ingots	No specified limits
CARLA	Induction	Carbon steel, stainless steel, aluminium, copper, lead (R&D)	3.2 t	Ingots, shield blocks, waste containers	Max. 200 Bq/g for beta-gamma nuclides. Max 100 Bq/g for alpha nuclides, separate limits for uranium
SEG	Induction	Carbon steel, stainless steel, aluminium, (planning to melt copper and titianium)	20 t	Ingots and shield blocks,waste containers and reinforcing steel	Normally <2mSv/h, greater dose rates with prior review and approval

9.4.2.5. Other decontamination techniques

In special cases, other decontamination techniques (*e.g.* laser, steam spraying, thermal erosion, pastes, gels, foams, etc.) have also been used in decommissioning or are in development. Some of them, however, require more or less complex application procedures or still need more development to allow industrial applications.

9.4.2.6. Selection of decontamination techniques for metals

To summarize a little bit the topic of metal components decontamination, a range of decontamination techniques can be used with a view to:

a. Reduce the dose rate of equipments or dismantling tools to be dismantled
b. Decategorization, this means to change the category of the radioactive waste
c. Reach residual contamination levels allowing to send the waste to a nuclear foundry with recycling of the materials in the nuclear industry or unconditional release after fusion
d. Reach residual contamination levels authorizing the immediate unconditional free release.

Table 9.8. gives a simplified overview of the techniques and their application field. There are several possibilities for each piece. The criteria used to choose a technique are mainly the following ones:
- the geometry and size of the pieces
- the nature and the level of the contamination
- the state of the surface and the type of material
- the availability of the process

One aims at minimizing the decontamination cost by choosing the best adapted technique. The goal is to carry out the decontamination in one single step.

Table 9.8. - Choice of the decontamination technique

Treatment	**Contamin.**	**Geometry**	**Material**	**State of the surface**	**Example pieces**
Polishing, grinding	External	Simple surface, rusted	CS; SS	Painted/non painted	Covers, flanges
Manual cleaning	External	Simple to complex	CS; SS	Painted/non painted	Beams, plates, cable runs, electrical cables, pneumatic part of valves
High pressure jet	External	Simple to complex	CS; SS	Painted/non painted	Tanks, radiators
Wet sand-blasting	External	Simple, large pieces	CS; SS	Painted/non painted	Beams, plates, tanks (after cutting)
Dry sandblas-ting	External	Simple, small pieces	CS; SS	Painted/non painted	Beams, plates, cable runs
Chemical decontamination	Internal	Simple to complex	CS; SS	Non painted	Tubes, valves, bends, tanks (after cutting)

Note: CS = carbon steel, mild steel
SS = stainless steel

9.5. Decontamination of buildings and concrete

For the decontamination of buildings in general and of all surfaces susceptible to be contaminated in particular, first the washing technique is used. This technique is very efficient for painted and slightly contaminated surfaces. Washing can be done manually for small surfaces, or by using an industrial washing machine.

For strongly contaminated surfaces and in particular for floors of which the coating is damaged, this technique is not aggressive enough. The contaminated layer has to be removed.

Simple processes, such as brushing, washing and scrubbing, and vacuum cleaning, have been widely used, since the need for decontamination/cleaning was first noted in the nuclear industry, and each nuclear facility has to some extent a certain practical experience of these kinds of decontamination processes. These processes are generally labour-intensive, but they have the advantage of being versatile. They are often used as a first step (*e.g.,* to vacuum dust and remove loose contamination) before or during dismantling, to prepare items for more aggressive decontamination using stronger processes.

Other, more aggressive techniques are grinding, spalling and drilling, high-pressure water jetting, foam decontamination, the use of strippable coatings, high-frequency microwaves, laser and induction heating. The use of most of these techniques is limited to specific applications in specific cases. Some of them have disadvantages such as spreading of contamination, or produce a lot of undesirable secondary waste. Some of them are also less suitable for industrial applications.
When decontaminating concrete surfaces, mainly mechanical scarifying techniques such as needle scaling, scabbling, or shaving, are used, although other "innovative" techniques are currently under development, like the laser ablation technique, micro-explosive decontamination, sand or abrasive blasting,.... Nevertheless, these techniques are either not yet completely industrialized or do not comply with all the requirements of a nuclear decontamination, and some of their parameters are still under analysis or development. Therefore, we will concentrate on the three major techniques, mostly used in various decommissioning projects worldwide:

- scabbling
- milling/shaving
- rock breaker / jackhammer

9.5.1. Decontamination by scabbling

Scabbling is a pure mechanical process. The upper layers of concrete are pulverized by means of tungsten carbide heads, that are moving up and down at high speed or rotating. The pulverized concrete is sucked by a filtration unit and collected in suitable packaging. The equipment available on the market are characterized by the number of heads, varying from 1 to 7. For places difficult to reach, an angle scabbling system can be necessary.
Both electrically and pneumatically-driven machines are available.

This technique is a dry decontamination method – no water, chemicals or abrasives are required. The waste stream produced is only the removed debris. Work rates are not easy to predict due to the variety of concrete composition and characteristics as well as to the different types of bits that may be used.
Scabblers are best suited for removing thin layers (up to 15 or 25 mm thick) of contaminated concrete (including concrete block) and cement. It is recommended for instances where:

- airborne contamination should be limited or avoided;
- the concrete surface is to be reused after decontamination;
- waste minimisation is envisaged;
- for instances in which the demolished material is to be cleaned before disposal.

The scabbled surface is generally flat, although coarsely finished, depending on the bit used. This technique is suitable for both large open areas and small areas.

9.5.2. Decontamination by milling/shaving

Milling is also a pure mechanical process. It concerns a tool composed of a series of diamond disks placed one next to the other, and the rotation provokes the erosion of the concrete till a depth of a few millimeters.

This machine is similar to a normal floor scabbling unit. It has a quick-change diamond-tipped rotary cutting head designed to give smooth-surface finish, easier to measure and ready for painting. It is capable of cutting through bolts and metal objects, which would have damaged the scabbling head of a traditional scabbler.

Based on the positive experience [9.2.] with the floor shaver a remote-controlled diamond wall-shaving system has been developed (Belgoprocess) as a solution for concrete decontamination of larger surfaces. The machine consists of:

- a remote-controlled hydro-electric power pack for the remote-controlled shaving unit;
- vacuum systems to fix temporarily vacuum pads holding the horizontal and vertical rails of the shaving unit;
- a simple xy-frame system containing a guide rail, a vertical rail and a carriage for the shaving head;
- a quick-change diamond-tipped rotary shaving head with dust-control cover for connection to existing dust-extraction systems.

The entire system is built up in sections which are portable by one operator. It removes a concrete layer in a controlled and vibration-free manner with the removal depth being controllable between 1 and 15 mm per pass, and producing a smooth-surface finish. The cutting head is designed to follow the contours of the surface being removed, and depth adjustments may be set manually in increments of 1 mm to minimise waste production. With 300 and 150-mm-wide shaving heads, both large areas and awkward corners may be accessed. When the vertical rail is fitted to the wall with the cutting head shaving, the horizontal rail may be disconnected and moved forward, thus ensuring continuous operation.

Production rates vary depending on the structure and the hardness of the concrete, the depth setting, the cutting speed and the type of diamond used. Heads can be used for shaving up to 2 000 m^2.

9.5.3. Decontamination with a rock breaker or hydraulic/pneumatic hammering

In nuclear facilities, the floors are generally more deeply contaminated than the walls and ceilings. Contamination depth up to a few tens of centimeters are commonly reported. For such contamination depths, the scabbling and milling techniques are often too slow and labour intensive. Therefore, one can also consider the use of a machine like the rock breaker or jack hammer, foreseen of a chisel, to decontaminate floors, and potentially walls when the contamination (or activation) reaches important depth.

Cutting and decontamination of concrete structures may be carried out with hydraulic or pneumatic hammers, either hands-on or using an electrically-powered, hydraulically-controlled support arm.

The latter may be equipped with a hydraulic hammer, an excavator bracket, or other tools, and is well suited for decontaminating floors and walls. A mini electro-hydraulic hammering unit (weighting only 350 kg) is commonly used in areas where contamination has penetrated deeply into the concrete surface, increasing the decontamination possibilities and reducing significantly the workload for the operators.

9.5.4. Comparison of various techniques and production rates

In order to compare the various presented techniques, one can use the production rate as an indicator. Nevertheless, like for the selection of decontamination processes for metals, other elements can influence the choice of a technique.
The table 9.9 gives some typical work rates [from 9.3.] obtained with the different kinds of scarifying techniques mentioned above.

Table 9.9 – Typical work rates obtained with different kinds of scarifying techniques

Scarifying technique	Layer thickness removed (mm)	Removal speed (m^2/h) (machine working time)
Needle scaler	2	0.1
Hand scabbler (1 head)	2	0.6
Floor scabbler (7 heads)	3	4.6
Wall scabbler (3 heads)	3	4.6
Wall scabbler (7 heads)	4	8.4
Floor shaver	1.5	13.6
Wall shaver	1.5	21

When selecting an appropriate decontamination technique for building surfaces, some general considerations should be taken into account. In any case, the use of techniques that would make contamination penetrate further into the substrate should be avoided. In addition, as general rules:

- For decontaminating painted floors and walls, where it may be proved that contamination has not penetrated into the substrate, simple processes as brushing, washing, scrubbing and vacuum cleaning may be used.
- For decontaminating concrete surfaces which are not painted and in which the contamination has slightly or more deeply penetrated the substrate, more aggressive techniques (e.g., scabbling, shaving, jack hammering or drilling) must be considered.

One could also be surprised not to find sand or abrasive blasting or water jetting as primary techniques to be used. This is intentional, as it has be frequently reported that these techniques, although sometimes efficient, present the risk of enhancing the penetration of the contamination into the concrete, ans spreading the contamination along the surface.

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Appendix 1

1. IMPLEMENTATION OF EXPERIENCE IN DECONTAMINATION OF POWER-UNITS OF NUCLEAR POWER PLANTS WITH RBMK REACTORS DURING DECOMMISSIONING [1]

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During the development of decommissioning of NPP units with reactors RBMK-1000 the reduction of radiation dose rate could be achieved, in particular, during dismantling of equipment of contour of multiple compulsory circulation (CMCC) and gaseous contour and provision of acceptable level of radiation safety of reactor installation during its preparation to the long-term conservation under control for account of carrying out the decontamination of this equipment. Moreover, because a significant part of equipment and systems of RBMK-1000 has only surface radioactive contamination (or non-radioactive at all), the metal after decontamination down to a level corresponding to the possibilities of limited or unlimited use may be used secondary, at least, for the needs of atomic energetic..

As result of decontamination of metal a significant reduction occurs of the amount of solid radioactive waste, but simultaneously appears a secondary, mostly liquid radioactive waste Because of this the development of highly efficient technologies of decontamination of metals (including "drastic" methods), giving the minimal amounts of waste (for examples, by means of use of decontamination by foam and vapor mixtures, gel, pastes, containing different chemical reagents), is an actual task.

For increasing the efficiency and reduction of volume of secondary radioactive waste before decontamination of concrete systems or types of equipment of unit with RBMK-1000 the levels and places of localization of radioactive contaminants has to be determined, as well as the further fate of materials and equipment to be decontaminated and, as consequence, the required extent of its decontamination.

At the present time practical experience is lacking on carrying out decontamination of equipment and systems of NPP with RBMK-1000 during decommissioning. Nevertheless, experience of decontamination of uranium-graphite reactors exists in Russia and relates mainly to decontamination during decommissioning of plants of first line of Beloyarsk NPP and industrial uranium-graphite reactors. At the same time works have been often realized on pre-restoration decontamination of CMCC units of NPP with RBMK-1000 during periods of planned-preventive and capital restoration

The problems arising during decontamination of CMCC of unit of NPP with RBMK-1000 are determined by specificities of its construction and exploitation. The construction of CMCC differs from the circulating contours of nuclear reactors of other type by larger volume (1200 m^3), more branching of communications, by presence of stagnant ("impasse") zones. These specificities of construction of CMCC does not allow to divide it on separate decontaminating sites, and this restricts the use of most of means of decontamination for the circulating loops of NPP.

For washing CMCC the use of single-bath or transforming subscribing of decontaminating solutions is recommended. The use of these solutions allows reducing the volume of liquid radioactive waste (LRAW) up to 5000-7000 m^3.

The selection of the subscribing of decontaminating solution in significant extent is determined by issue of construction materials of CMCC, amount and chemical composition of corrosion deposits, level of radioactive contamination of sites of contour and radiochemical composition. The stainless steel 08X18N10T and alloy of Zirconium 110 are the main construction materials of CMCC.

The water-chemical regime of CMCC is the neutral un-correction one, the content of oxygen in the coolant on some sites approaches 100-200 μg/kg [2], that is why the distinctive specificity of composition of corrosion deposits in the contour is the lower content of chromium (because it oxidizes into soluble Cr (IV)) and a significant share of hematite (α-Fe_2O_3) in porous deposits.

The phase content of deposits of products of corrosion on separate sites of CMCC is different : in samples from drum-separator – 54% of magnetite (Fe_3O_4) and 46% magemite (γ-Fe_2O_3); in system of acid vapour - 50% of magnetite and 5% of magemite; in de-aerators - 15-28% of magnetite, 42-85% of hematite and 0-30% of magemite. The composition of dense oxide film of stainless steel corresponds to the spinel $[Fe,Ni]O \cdot [Fe,Cr]_2O_3$ [3].

In corrosion deposits, oxides of iron with admixture of nickel, chromium and manganese dominate. Also up to 5% of silicon, might be present in the composition of the deposit, which appears to be due to filters with perlite in the system of by-pass clean-up, the concentration of silica acid in water of CMCC obis is present in the range 70-400 μg/l. It is necessary to notice that the formation in the contour of silicon-containing deposits seriously complicates decontamination.

The overall amount of deposits of products of corrosion in CMCC approaches 1000-13000 kg. The specific activity of porous deposits is $3x10^{-8}$-$4x10^{-3}$ Ci/g (or 1-7,5 microCi/cm^2), the overall activity of corrosion deposits in CMCC - $\geq$ 500-1000 Ci. Up to 90-95% of overall radioactivity is concentrated in porous deposits. Furthermore the input of activity of surface of CMCC in total activity of products of corrosion is not more than 0,5-1%.

The radioactivity of deposits of products of corrosion is determined in general by radionuclides such as ^{51}Cr, ^{60}Co, ^{95}Zr, ^{95}Nb and by products of fission. The input of some radionuclides in dosimetric situation is changing depending on the period of accumulation of deposits - increasing the input of ^{60}Co and decreasing the input of ^{51}Cr and ^{58}Co [3,4].

The accumulation of corrosion deposits resulted in increasing of dose exposure, in particular, in the under-control room – up to 3600-14000 mR/h (after shut-down of reactor), in the premises of GNC – form 50 till 2500 mR/h.

For decontamination of CMCC of units of NPP with RBMK-1000 a few methods have been developed: chemical, low-reagent and un-reagent.

In the chemical method the single-solution (single-bath) decontamination subscribing, with possible introduction of additive reagents in course of the process a (transformation of solution) [3].

The first developed variant of analogous subscribing was the decontamination solution on the basis of oxalic acid (10-20 g/l) [5]. For bringing pH up to 2,0-2.5 in the beginning a solution of ammonia has been used, and thereafter of potassium hydroxide, because difficulties arise with the treatment of liquid RAW and cleaning up of condensate from ammonium. The treatment by the abovementioned solution is carried out during 9-15 hours at 90°C.

Then a solution of hydrogen peroxide is injected into the contour up to a content 2-5 g/l, the treatment is continued during 5-8 h (70-90 ^{0}C), thereafter the solution is drained or is cleaned up from the radioactive products of corrosion by means of a by-pass clean up. After the drainage of the solution the CMCC is washed by water. Achieving more efficient clean up of surfaces of CMCC 2 cycles of chemical decontamination is possible.

The main disadvantage of this technology is the significant liberation of radioactive Iodine in the environment during injection of H_2O_2. The formation of volatile forms of radioactive Iodine is due to oxidation by hydrogen peroxide of ion I^- up to molecular iodine in acid media; moreover up to 30% of Iodine is transferred to the vapor phase together with gaseous products of oxidation of oxalic acid and decomposition of hydrogen peroxide. The liberation of gases in this technology is of the order of 3000 m^3 (composition: CO_2, O_2, H_2) [5].

For catching radio-iodine in the system of vacuum evaporation of the contour, two coal absorbers AUI-1500 are used; for preventing the accumulation of hydrogen a continuous scavenging of separators by vapor is carried out by compressed air with a consumption 5000 m^3/h.

To reduce liberation of radio-iodine and gases an improved method of decontamination of CMCC has been developed by solutions of oxalic acid (5-10 g/l) with addition of small concentrations of KNO_3. After the treatment by this solution at 90-100°C during 25-50h the solution is partially drained and H_2O_2 is injected; moreover the concentration of $H_2C_2O_4$ and KNO_3 is reduced approximately by a factor 10. The treatment by this solution is performed during more than 3-8 h at 70-75 ^{0}C. Then the solution is cleaned up on the filters of bypass and is drained. During implementation of this technology the gas liberation is reduced up to 200 m^3, and the consumption for scavenging of separators amounts up to 500 m^3/h. This method of decontamination is called "nitrox".

Both described methods have been used as staff methods of pre-restoring decontamination at Leningrad, Chernobyl and Kursk NPP. The decontamination of CMCC by this method has been carried out frequently, starting in 1976, it has been applied till now. The efficiency of these methods of decontamination is approximately the same, and this is illustrated by the data given in table 1.

The decontamination of CMCC and can be carried out together with that of active zones, moreover the non-hermetic thermo-emitting assemblies (TEA) are removed, and the staff equipment is used for the circulation of solutions.

At interception of active zone the circulation of solutions could be carried out following different schemes:

1) From the pressure collector of main circulation pumps (MCP) towards the pumps of cooling and through the crosspiece between the supply and return of scavenging on the system of special water cleaning up (SWC-1) to the system of return of scavenging and on the loop of nutrient water to the drum-separator. Then to the pressure collector GCN, either through the down-comer pipe-lines and connector between the suction and pressure of GCN, either through the technological channels, low water communications (LWC), the distributing collector GCN.

2) From the pressure collector of GCN through the NVK, technological channels, vapor-water communications (VWC), drum-separator, down-comer pipe-lines to the suction collector GCN.

The removal of radioactivity from the contour is carried out at partial draining and displacing of decontamination solution, as well as for account of cleaning (around 5% of total activity) on perlite filters of system of bypass cleaning.

The part of the insoluble radioactive suspensions in the process of circulation of solution is precipitated in the stagnant zones of CMCC, accumulated in the gap between the ball-throttle flow meter (BADF) and the vessel of NVC, in the up-dumping nipples of PVC and internal connectors of drum-separators, nipples on the suction collectors GCN. For improving the removal of non-solubilized active suspensions of products of corrosion it is necessary to change the scheme of water return, setting it from the suction GCN to the bypass cleaning-up, or to foresee the possibility to change the scheme of solution circulation.

The small-reagent decontamination of CMCC includes disaggregating particles of corrosion deposits by injection of 25-100 mg/l of ions NO_3^- or NO_2^- in water of contour or by saturation of water by carbon dioxide. As a result, the increasing is occurring in 2-4 times of the content in water of suspension of active products of corrosion. Then the suspended products of corrosion are removed from the coolant in the system of bypass clean-up of water of CMCC [6].

Table 1 - Parameters and efficiency of chemical decontamination of CMCC of unit with RBMK-1000

Unit	Data	Composition of solution	Premises	K_d on dose rate	removed Activity, Ci	Fe, kg	Prolon-gation h
LAES-1	1976	$H_2C_2O_4$+ NH_4OH+H_2O_2	Under-control rooms GCN	2,1 2,7-3,2	10000	936	~ 100
LAES-2	1977	The same	Under-control rooms GCN Drum-separators	2-3,6 2,2-9,5 5,7-4,4 (av. 4,6)	6000-7000 (LRAW - 4300 m^3)	1300	95
LAES -1	1989	$H_2C_2O_4$+ KNO_3+H_2O_2 ("nitrox") 2 cycles	Under-control rooms GCN Premises NVC Drum-separator	18-32 2,9 2,1 10,1 (av. 6,3)	5000	1000	118
LAES-3	1995	The same 2 cycles	Premises. NVC Boxes GCN Boxes GCN (connector NC-VC) Downcomer- pipe-lines Drum-separators Under-control rooms	1,0 (contam. floor) 1,4 52 1,0-3,6 (av. 1,8) 1,0-11 (av. 5,75) 12-24 (av. 18)	1430	900	160

The un-reagent decontamination is based on the changes of the charge of particles of products of corrosion and on the decontamination during the cooling of reactor and during injection of 5-7 mg/l of air oxygen in the water of contour.

Un-reagent oxygen decontamination is realized by means of replacement of water of CMCC cooled up to 40-50 ^0C by the chemical desalinated water, saturated by oxygen of air (up to 8 mg O_2/l) [7]. The replacement of water can be carried out by means of displacing of water of CMCC. The injected desalinated water is mixing during 2 h., and then the time-lag during 10 h is carried out at natural circulation and then the removal of products of corrosion on the bypass cleaning during 50 h. The water supply on the replacement is realised at scheme: the collectors of system of accidental cooling of reactor (SACR) – the technological channels – the active zone – the drum-separators – the down-comer pipe-lines – the drainages of GNC. The un-reagent decontamination have been carried out in 1984 y. on 2-d и 3-d units of Leningrad NPP, I-st and 2-d units of Chernobyl NPP, in 1987 y. – on the 2-d unit of Chernobyl unit of NPP.

The carbon dioxide decontamination includes the cooling of coolant up to 60-100°C, its saturation by CO_2 up to content 100-400 mg/l, mixing, time-lag and cleans up from the products of corrosion [7]. The total prolongation of the process is 75-92 h. The carbon dioxide decontamination has been carried out in 1984 y. on 3 units of the Chernobyl NPP.

The results of implementation of un-reagent and carbon dioxide methods of decontamination are given in the table 2.

In total, as result of carrying out if non-reagent or low-reagent decontamination the reduction of dose rate in the buildings of CMCC is occurring in average in 2-4 times, furthermore from 150 till 670 Ci of activity is removing from the contour.

The efficiency of un-reagent method of decontamination is compared with chemical one. The intermittent of the process, the minimal corrosion of materials, the decreasing of amount of liquid RAW in 3-6 times in comparison with chemical decontamination have to be also mentioned as the advantages of un-reagent technology. All operations are included in the staff regulation of NPP with RBMK-1000.

In comparison with un-reagent decontamination the carbon oxide decontamination is more technological, furthermore the liquid radioactive waste are practically absent.

It is necessary to notice that during decommissioning of NPP non-staff, e.g. sufficient drastic methods and procedures of decontamination could be used.

Nevertheless their selection requires also precautions and optimization, because the process of intense solubilization of metal to be decontaminated metal may cause an increase of the content of iron and other components of construction steels in the secondary liquid waste, what can complicate further treatment.

Finally, on the base of existing experience about decontamination of equipment and KMCC of RBMK-1000 units one may say that there is sufficient existing methods and technologies for decontamination, that is why the problems of decontamination of KMCC, described here above, could be solved during decommissioning of these units.

This experience and these technologies of decontamination could be used in the future with success during decommissioning of the first unit of Leningrad NPP. Nevertheless in the case of large scale development of works on dismantling of equipment, the amounts and nomenclatures of technical means of decontamination existing at the present time on NPP with RBMK-1000, may be not sufficient for fulfilling the required volume of different decontamination works.

Table 2 - The efficiency of decontamination of CMCC of unit with RBMK-1000 by non-reagent and carbon dioxide

Unit	Date	Dose rate, μR/s			K_d	Removed		Prolongation, h
		Premises	before decont.	After decont.		Activity Ci	Fe, kg	
LAES-2	04. 1984	Under-control rooms	350; 420; 350; 650	180;230; 160;180	2,4	670	-	69
		NVK	88; 146; 56; 186	70; 140; 30; 48	2,0			
		Boxes GCN	53; 30; 146; 127; 47; 29; 103	43; 47; 42; 28; 20; 23; 28	2,46			
		Drum-separators	24; 18	18; 19	1,14			
		Downcomer-pipelines	7;3; 6,3	6; 0; 2,6	1,82 av.2,0			
ChNPP-2	06. 1984	Under-control rooms	600	80-110	6,4	427	2,63	64
		Boxes GCN	130;70; 320;417	18; 24; 17; 27	10,4			
		NVK	221; 158; 270	199; 100; 183	1,28			
		Drum-separators	98	93-100	1,05			
		Downcomer-pipelines	50; 64 415; 377	14; 56 105;133	2,4 3,38 av.4,2			
LAES-3	05. 1984	Under-control rooms	405; 335; 337; 387	100; 80; 84; 83	4,2	227	2,33	63
		Boxes GCN	85; 127; 121; 83; 124; 138; 271; 66	66; 159; 83; 78; 181; 130; 261; 47	1,1			
		NVK	119; 188; 148; 225	78; 128; 102; 130	1,55			
		Drum-saparators	25; 22	19; 18	1,28			
		Downcomer-pipelines	8; 7	4; 4	1,28 av.1,9			
		Drum-separators	27,5	30	1,0			
		Downcomer-pipelines	5,5	4,2	1,4 av.2,26			

In connection with this during development of project of decommissioning of the first unit of Leningrad NPP it seems rational to determine the need in these means and their nomenclature as well as the volume of required research and construction works, in particular, regarding the use of desolate technologies for decontamination works.

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2. DECONTAMINATION OF EQUIUPMENT AND SYSTEMS OF LENINGRAD I-ST UNIT NPP DURING DECOMMISIONING [1]

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The presently considered strategy of decommissioning of the first unit of Leningrad NPP with reactor RBMK-1000 includes: preparation of unit to decommissioning (duration about 5 years), preparation to the long-term safety conservation (up to 5 years), long-term safety conservation under control (not less then 50 years) and final stage of dismantling of reactor constructions (from 2 to 5 years).

During the preparation and development of works on decommissioning of this unit the main way of reducing radiation dose rate will be the decontamination of equipment, systems and constructions, aiming at a suitable level of radiation safety of systems and equipment and, in the first turn, of reactor installation as well as during preparation to long-term conservation under control.

Some problems, which will arise during decontamination of reactor systems and equipment during decommissioning of this unit, are considered in this contribution.

On the first stage of preparation of unit to decommissioning after final shut down and removal of the nuclear fuel, a complex inspection of unit, has to be carried out, in particular, about the state of the equipment, the systems and construction of reactor installation.

As a result of this inspection, the levels of radioactive contamination of technological contours, systems and equipment of reactor installations has to be determined, as well as the radiation dose rate in the control-rooms and other places of provisional dismantling works.

Later on this stage the following problems have to be solved taking into account the results of this inspection:
- the rationality of carrying out the decontamination of contour of multiple circulation (CMCC) of the coolant;
- the necessity of decontaminating technological channels and gas contour;
- the selection of efficient methods of decontamination of systems of reactor and equipment to be dismantled;
- the necessity to develop and create technical means of decontamination, and in case of necessity, of technological instrumentation and robot-automatic devices required for a possible remote control of decontamination;
- the preparation of sites of treatment for initial and secondary radioactive wastes (RAW);
- the evaluation of required expenses and material-technical provision to decontamination.

The analysis of the first three abovementioned problems, as well as the possible solutions are given hereunder.

After completion of staff operations on removal of thermo-emitting assemblies (TEA) from the active zone of reactor further operations have to be carried out on removal of spent TEA from the near-reactor decay tanks (DT) of fuel and its transportation to the plant storage of spent nuclear fuel (SSNF), and parts of TEA, suitable for further use, - in the DT of acting units. In the next case TEA, suitable for reuse, have to be cleaned up on a special installation from the radioactive corrosion deposits by mean of appropriate ("soft") decontamination solutions.

In turn, the near-reactor DT after liberation from the TEA have to be cleaned up from mechanical and radioactive contaminations and, in case of possible re-use, for example as a place for organised storage of solid RAW, they have to be restored (with liquidation of possible leaks) and put in working readiness.

The operations on clean up and decontamination of near-reactor DT have to include the follow stages: removal from the bottom of DT of elements of destructed TEA, other subjects and sludge deposits, the decontamination of walls and bottoms of DT.

The procedure of decontamination of DT may be realised, for example, by following exploring methods: by means of flow-jet treatment by chemical solution, by means of hydro-monitor or electrochemical method using remote controlling taking out electrode, the shifting of which along the walls of DT is realised following a prescribed programme.

In the case of selection of flow jet chemical decontamination as a main method for decontaminating the of walls of DT electrochemical decontamination could be also used later on, aiming at final decontamination of welded links of steel fettling of DT.

After fulfilling the operations on removal of TEA from the unit and next clean up of DT the stage of preparation of first unit of Leningrad NPP to the decommissioned can be considered as completed.

The next stage deals with the preparation of the unit to long-term safety conservation. Analysing the necessity of decontamination of CMCC will be the main problem, requiring analysis and solution. Solving this problem will depend mainly on the levels of dose rate in the premises of CMCC and on the necessity to carry out works shortly in these premises.

The experience of exploitation, carrying out restoration and reconstruction of units of NPP with reactors RBMK-1000 shows, that the dose rate of radioactive exposure in control-rooms of CMCC during works of reactor is 100 - 200 μR/s. In the process of reduction of thermal power of reactor and/or shut down the dose rate increases up to 1000 - 2000 μR/s. Moreover in other premises of CMCC the gamma-background is also increasing, what deals with precipitation of high level radioactive products of corrosion and the inferior elements of the constructions of CMCC during the decrease of circulation of coolant.

Nevertheless already in the conditions of preparation of unit for decommissioning, when the nuclear fuel is removed from the reactor, as shown by the experience of restoration and reconstructions of RBMK-1000, the activity of CMCCK is reduced by a factor of 100 one year after shut down, and 5 years after final shut down the radioactivity of deposits in the contour is additionally decreased by not less than 4 - 5 times (mainly Co-60). The gamma dose rate in the control-rooms and other premises of CMCC is also decreased as a consequence of radioactive decay of most of radionuclides from corrosion (with the exception of Co-60) and of the absence of transfer of radioactive deposits from the active zone.

In connection with this, during the preparation of the unit to a long-term safe conservation the necessity of carrying out the decontamination of CMCC has to be evaluated from the point of view of acceptable dose rate for personnel at the time of start of execution in the premises of contour.

Nevertheless taking into account the experience of planned and capital restorations of units of NPP with reactors RBMK-1000 the carrying out of decontamination of CMCC in assemble on the stage of preparation of unit to the long-term safe conservation seems rational, because in this case a significant part of corrosion deposits, will be removed from the CMCC, having a porous structure and, as consequence, the radiation situation in the premises of contour will be improved.

On this stage of decommissioning of units with RBMK-1000 decontamination of CMCC the methods could be selected on the base of un-reagent (or small-reagent) or chemical decontamination.

The un-reagent decontamination may be realised by replacing cooling up to 40 - 50 °C coolant of CMCC by desalinated water, saturated by air oxygen (6 - 8 mg/l of oxygen) [2]. After this, the desalinated water is mixed, отстаивается and then active products of corrosion are removed on bypass clean-up. The replacement of desalinated water may be carried out by on-line displacing: the collector of system of accidental cooling of reactor (SACR) – the distributing-group collectors – technological channel of active zone – the drum-separator – suction collector of main circulating pumps (MCP) – drainages MCP.

The advantage of selecting un-reagent technology are the minimal amount of liquid radioactive wastes, the use of only staff equipment of CMCC; moreover all operation are included in the staff rules of NPP.

If chemical methods are used for the decontamination of CMCC the decontamination procedure will include the following operations [3]:

- injection of solution of potassium nitrate and then oxalic acid up to a concentration 3,0 - 3,5 and 9 - 10 g/l appropriately to the water of CMCC;
- circulation of the solution for 20 h at temperature 90 - 100 °C;
- partial draining of solution with replacement on water and injection of hydrogen peroxide up to a concentration around 0,25 g/l;
- circulation of solution during 3 - 5 h at temperature of 70 - 75 °C;
- drainage of decontaminating solution and scavenging by water.

The data for the comparison of efficiency of these methods of decontamination are presented in table 1 below.

From these data, the method of chemical decontamination is characterised essentially by more efficient removal of iron and gamma-activity. And, in that case, most of the radioactive deposits are removed from the active zone. In the case of utilisation of un-reagent decontamination, are removed mainly the deposits from the surfaces of CMCC.

Nevertheless, by comparing coefficients of decontamination, it is clear that as concerns the volumes of liquid RAW, as well as the technological characteristics of these methods of decontamination, un-reagent method of decontamination on this stage of decommissioning is more favourable, especially taking into account the removal of TEA from reactor on the previous stage.

After these procedures of decontamination, the water from CMCC is drained and the contour is dried. The equipment, which is not required during decommissioning of unit, may be dismantled according to the decrease of its radioactivity.

As mentioned, the solution of problem of rationality of decontamination and dismantling of technological channels requires special considerations at the stage of preparation of the unit to a long-term safe conservation under control.

It is known that during exploitation of RBMK-1000 together with activation of technological channels radioactive corrosion deposits are formed on their surface. Then, if during the preparation of the unit to a long-term safe conservation, the decontamination of CMCC has been already carried out, then the layer of porous corrosion deposits from the technological channels will be removed. If the decontamination of CMCC has not been carried out then the technological channels will need to be decontaminated after removal.

Nevertheless the question about dismantling of technological channels is more rational to consider after decontamination of CMCC. This deals with the fact, that as was the case with the industrial uranium-graphite reactors, given radiation changes during the exploitation of reactors, given damages to the conformity of technological channels, leaks of water and vapour to the graphite casing, the graphite may be submitted to partial destruction in the case of removal of technological channels.

Table 1 - Comparison of characteristics of efficiency of un-reagent and chemical decontamination of CMCC units of NPP with RBMK-1000

The title of premises or equipment	Initial dose rate Doses, $\mu R/s$	Coefficient of decontamination (Kd)	
		Un-reagent decontamination	Chemical decontamination*)
Control-rooms (016/1,2)	335 - 970	4,2	18-32
Premises of low water communications (033/3,4)	36 - 119 166 - 240	3,1	2,1
Boxes GCN (08/9-16)	25 - 125	3,9	2,9
Down-comer pipe-lines, suction collector GCN (115/3,4)	7 - 20	1,5	10,1
Drum-separators (505/3,4)	21 - 140	1,3	6,5
Average Kd	-	2,0 - 4,1	1-st cycle - 2,6 2 cycle - 6,1
Reduction of dose rate	-	in 2,0 - 4,1 times	in 5,2 times
Removed: Gamma-activity, Ci Iron, kg	-	 142 - 670 2,3 - 2,6	 5000 580 + 390
Volume of Liquid RAW , m^3	-	1200 - 1800	4000 - 5000
Prolongation of the process, h	-	60 - 70	1-st cycle - 62 2-d cycle - 56

*) – after two cycles of decontamination

The long-term exploitation of units of Leningrad NPP demonstrated that in general the state of the graphite laying is satisfactory and the technological channels may be removed. Nevertheless, the study of graphite laying shows that it is deteriorated, there is a radial shrinking of the graphite cells (up to 1,5-2,3 mm), caused by thermal and radiation action. The shrinking in the graphite columns is irregular, the places of maximal shrinking are located in the zones of highest neutron flux: the average rate of reduction of internal diameter of holes in the graphite units is 0,09 mm/year [3]. The reduction of size of the holes in the graphite units together with the deformation of zirconium channels and the increase of its diameter with a rate around 0,1 mm/year results in a decrease of size of the gap and an increased stress during removal of the technological channels. The reduction of height of the graphite columns occurs at a rate 3,9 mm/year (calendar), in some cases the graphite crumb, fragments of graphite rings, sometimes spallation fragments have been found along the height of laying.

In case of close contact of graphite units and technological channels the limit of strength of graphite may be achieved after 20-25-th years of exploitation. Then cracks occur together with structural degradation of graphite, as well as wedging of channels.

For preventing such phenomena on the units of Leningrad NPP a reconstruction is carried out on the active zone of the reactors with restoration of a correct size of gaps [4].

In connection with this, depending on the state of the graphite laying which is determined by the duration and intensity of neutron exposure, by the reduction of the integrity of technological channels and the possibility of oxidation and radiation swelling of graphite, the following two alternatives have to be evaluated:

- in case of a normal state of graphite laying after decontamination of CMCC, the technological channels are removed and then undergo additive procedure of electrochemical decontamination (it is necessary to notice, that for the electrochemical decontamination the electrochemical bath, existing in the central hall of reactor, may be used, nevertheless carrying out electrochemical decontamination of technological channels requires additive treatment);
- in the case of a very bad state of the graphite laying the technological channels are not removed for a period of long-term safe conservation under control.

Nevertheless the final solution about the rationality of removal of technological channels at transfer of unit to the stage of long-term safe conservation under control have to be determined during the development of the project on decommissioning of unit taking into account the consideration of all abovementioned and other parent factors.

The gas contour has to be the next object of decontamination on this stage of decommissioning; nevertheless it is necessary to notice that this contour cannot be decontaminated by chemical methods.

In connection with this, either the equipment of gas contour has to be dismantled and then decontaminated, either it has to be decontaminated in assembly by scavenging by gas or vapour admixtures or by means of thermal method.

If necessary, some places may be localised and conserved (for example, the place of condensate-nutrient channel after carrying out of «soft» chemical decontamination).

During dismantling of radioactive systems and equipment the problem of their pre-dismantling decontamination may be solved by means of polymer covers of different purpose [5].

At the present time a few classes of such polymer covers have been developed: protecting ones – for the protection of metal from corrosion; decontaminating ones – for the removal of radioactive contaminants from the surface; isolating ones – for the protection of surfaces from the radioactive contamination or isolation of contaminated surfaces; localising ones – for isolation of local contaminants, spills etc.

The last two from the above-mentioned classes of polymer covers may include strip-coats, as well as outstrip-coats. From the point of view of fire safety the use of water-soluble compositions for example, based on polyvinyl alcohol (PVA) is more rational.

The polymer covers may be used for isolation of contaminated surfaces of equipment as well as during its cutting and partition, and also for the protection of clean surfaces, with a goal of prevention of possible spreading of radioactive contaminants to the environment during transportation of dismantled radioactive equipment to the place of burial or temporary storage.

For analogous goals PVC polymer compounds have been used. In this case the thickness of the cover is about 100 - 250 μm, at which the mechanical strength of the cover is conserved during 1 - 1,5 years. These polymer covers may be applied onto the surface of silicate materials, for example, on concrete and bricks.

If necessary, the polymer cover may be removed after its treatment by water or by vapour-water mixture. For improvement of stripping ability of polymer films, the covering surface of equipment may be preliminary treated by an alkaline solution.

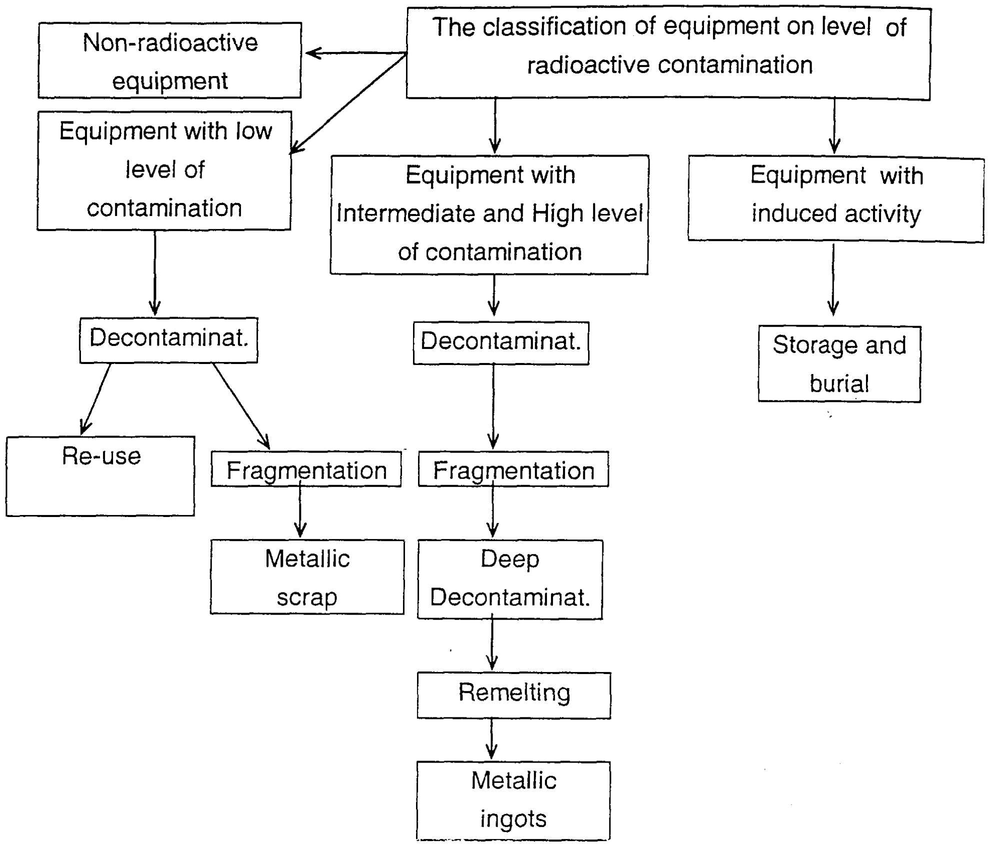
A few compositions of PVC decontamination covers have been developed which may be used for different covers, including that with a complex relief, which include decontaminating agents - organic or mineral acids.

Depending on radionuclide composition of surface contaminants the coefficients of decontamination by means of these polymeric covers approach 8 - 130. The advantage of these decontaminating covers deals with the fact that its implementation practically excludes the formation of secondary liquid radioactive waste. The scaled polymer films may be gathered, crushed and then compacted, for example, by methods of bitumisation or hot pressing.

If on-site decontamination is impossible or irrational, depending on the level of radioactive contamination, this equipment may be dismantled and sent to decontamination to specialised sites.

In general, the procedure of decontamination of dismantled equipment may be realised following the scheme given on picture 1. From the picture, one sees that the operations on decontamination allow to recycle the part of decontaminated equipment and metallic constructions, for reuse in the national economy and to reduce the volumes of RAW, arising during decommissioning.

In conclusion one may notice that at the present time there exist a complex of technologies and technical means for the decontamination: the accumulated experience of works on decontamination of different systems and equipment during exploitation of units of NPP with RBMK-1000 allows solving most of the problems of decontamination during decommissioning of these units.



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