出國報告（出國類別：開會）

出席 2018 年太平洋盆地核能會議

服務機關：台灣電力公司

姓名職稱：黃平輝專業工程師

派赴國家/地區：美國/舊金山

出國期間：107 年 9 月 29 日至 107 年 10 月 7 日

報告日期：107 年 11 月 23 日

行政院及所屬各機關出國報告提要

出國報告名稱：出席 2018 年太平洋盆地核能會議

頁數 50 含附件:■有□否

出國計畫主辦機關/聯絡人/電話：台灣電力公司/陳德隆/(02)23667685

出國人員姓名/服務機關/單位/職稱/電話：

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出國類別：□1 考察□2 進修□3 研究□4 實習■5 其他 –出席國際會議

出國期間：自 107 年 9 月 29 日至 107 年 10 月 7 日 出國地區：美國

報告日期：107 年 10 月 23 日

分類號/目

關鍵詞：太平洋盆地核能會議

內容摘要：(二百至三百字)

本次出國之任務為出席民國 107 年 9 月 29 日至 10 月 5 日在美國舊金山舉行之 2018 年太平洋盆地核能會議。於會議中發表「核二廠護箱裝載池安裝格架案之安全分析」(Safety Analyses for Installing Fuel Storage Racks in Cask Loading Pool at Kuosheng Nuclear Power Station)論文。此論文重點在介紹本公司(台電)在「核二廠護箱裝載池安裝格架案之安全分析」的一些關鍵的安全議題與重要的經驗回饋，期能與國際核能社會分享國內的創新技術與重要經驗。

核二廠一號機於 105 年 11 月大修時即面臨用過燃料池貯存容量不足,無法繼續運轉困境，本公司經探討各種可行性方案，採用護箱裝載池安裝 4 組龍門電廠二號機庫存格架的的創新作法。每部機組可增加 440 束用過燃料貯存空間，可讓機組繼續運轉 3 年。本案依據原能會所發布審查規範之要求提出安全分析報告，報告內容包括 5 大技術範疇(臨界安全、熱流分析、結構分析、輻射安全、事故評估)之分析與評估結果，另針對相關議題亦提出詳盡的評估與說明。本案原能會審查極為嚴謹與詳盡，在參與人員共同努力下，原能會所提出的關鍵安全議題及時獲得解決，本案於 106 年 4 月 6 日順利取得原能會核准，一號機在完成燃料挪移等作業後於 6 月 9 日併聯、6 月 16 日滿載發電，及時因應夏季尖峰用電需求。在綠能尚屬發展階段、夏季多次亮出「限電警戒」紅燈 (備轉容量低於 90 萬瓩)的嚴峻情況下，本案順利推動對台灣避免限電有關鍵性的貢獻。

摘 要

2018 年太平洋盆地核能會議(第 21 屆 PBNC)係太平洋核能理事會(Pacific Nuclear Council，簡稱 PNC)每兩年辦理之國際會議，本次由美洲核能協會(American Nuclear Society)負責籌備。

會議之主題為「核能的永續與促進」(Sustaining and Advancing Nuclear Energy)，有美國、加拿大、日本、澳洲、法國、英國、墨西哥、阿拉伯聯合大公國、沙烏地阿拉伯、中國大陸、我國等 12 個國家超過 300 位的專家代表出席，發表的論文約有 140 篇。除了專題研討論壇(Plenary Sessions)以及技術分組會議，2018 年會議還設有 36 個贊助公司的技術展示攤位。

我國出席人員包括原能會三名(核管處許明童、核技處戈元、物管局李彥良)，台灣電力公司一名(黃平輝)，清華大學一名(李敏院長)及高雄大學一名(張惠雲副教授)。分別參加各項議程及發表論文，並與各國專家學者交換意見，瞭解目前各太平洋盆地核能國家核能發展現況、新一代核能技術及核能安全等相關議題，並與與會各國專家學者交換技術意見。

目　　　錄

一、目的

本次出國之任務係出席民國 107 年 9 月 30 日至 10 月 5 日在美國舊金山舉行之 2018 年太平洋盆地核能會議(PBNC)，PBNC 是在太平洋核能理事會(PNC)推動下，由太平洋沿岸國家的學術團體輪流舉辦的國際會議，從 1976 年開始，每兩年召開一次。PBNC 大會的宗旨是促進核能技術的交流，爲太平洋地區核能的開發和全世界核能事業的發展作出貢獻。由於太平洋地區是世界核能發展最爲活躍的地區，故 PBNC 為世界核能界之盛事。

本次會議由美洲核能協會(American Nuclear Society)負責籌備。會議之主題為「核能的永續與促進」(Sustaining and Advancing Nuclear Energy)，有美國、加拿大、日本、澳洲、韓國、法國、英國、墨西哥、阿拉伯聯合大公國、沙烏地阿拉伯、中國大陸、我國等 12 個國家的專家代表出席，發表的論文約有 140 篇。除了專題研討論壇(Plenary Sessions)以及技術分組會議，2018 年會議還設有 36 個贊助公司的技術展示攤位。

我國出席人員包括原能會三名(核管處許明童、核技處戈元、物管局李彥良)，台灣電力公司一名(黃平輝)，清華大學一名(李敏院長)及高雄大學一名(張惠雲副教授)。

於會議中發表「核二廠護箱裝載池安裝格架案之安全分析」(Safety Analyses for Installing Fuel Storage Racks in Cask Loading Pool at Kuosheng Nuclear Power Station)論文，參加各項議程並與各國專家學者交換技術意見，同時於會議中蒐集並向相關專家洽詢本公司需要釐清議題之資料(如除役期間保安法規及做法)。

二、過程

(一)行程

此次奉派出國，行程如下表：

日 期	行程	工　作　內　容
9 月 29 日	台北－美國舊金山	往程
9 月 30 日至 10 月 5 日	美國舊金山	出席 2018 年太平洋盆地核能會議
10 月 6 日至 10 月 7 日	美國舊金山－台北	返程

(二)出席 2018 年太平洋盆地核能會議

2018 年太平洋盆地核能會議(第 21 屆 PBNC)係太平洋核能理事會(PNC)每兩年辦理之國際會議，本次會議由美洲核能協會(American Nuclear Society)主辦，係美國第五次承辦 PBNC。另加拿大原子能公司(Atomic Energy of Cannada Limited, AECL)重組成立的新機構「加拿大核子實驗室」(Canadian Nuclear Laboratories)最近聘請很多美國管理人員，為了發展國際業務，也積極參與本次會議之籌辦，專題研討論壇主席(Kathryn McCarthy)及技術議程主席(Corey McDaniel，原任職美國能源部)都由加拿大核子實驗室人員擔任。

會議之主題為「核能的永續與促進」(Sustaining and Advancing Nuclear Energy)，從管制、部會、產業、未來展望、最新科技及應用等不同觀點進行研討。有美國、加拿大、日本、澳洲、韓國、法國、英國、墨西哥、阿拉伯聯合大公國、沙烏地阿拉伯、中國大陸、我國等 12 個國家的專家代表出席，經大會統計共有超過 300 人參加，發表的論文約有 140 篇，每天有上午半天的專題研討論壇及下午半天的技術分組會議。本次會議之主要活動內容摘述如下：

1.報到及歡迎晚會

大會在 9 月 30 日(週日)辦理報到及舉辦歡迎晚會,歡迎晚會由美國愛達荷州參議員 James Risch 對與會者致歡迎詞，以及進行「國會主題」(Congressional Keynote)演講，作為美國核能研發重鎮-愛達荷國家實驗室所在地選出的參議員，Risch 積極表達對可靠、清潔的下一代核反應器的支持，並明確指出下一代核反應器將成為美國能源戰略的一部分。

2.開幕及主題演講

在 10 月 1 日(週一)上午八時，美洲核能協會會長 John Kelly 代表 ANS，PBNC 大會專題研討論壇主席、加拿大核子實驗室(Canadian Nuclear Laboratories)研發副總兼實驗室主任 Kathryn McCarthy 代表加拿大核子實驗室,以及 PBNC 大會名譽主席 Mark Peters (愛達荷國家實驗室主任)對各國與會者表達歡迎之意，接著是當日的「主題演講」(Keynote)及「小組討論會」(Panel)：

(1)政府觀點主題演講

「政府觀點」(Government Perspective)分為管制及部會等兩個主題,「管制觀點」由大會邀請貴賓美國核能管理委員會主席 Kristine Svinicki 及加拿大核能安全委員會(CNSC)管制業務主任 Ramzi Jammal 分別主講美國及加拿大的重要核能管制事項;「部會觀點」由美國能源部核能高級顧問 Suzanne Jaworowski 及加拿大能源部門電力資源處(Electricity Resources Branch, Energy Sector)主任 Marco Presutti 主講重要核能發展事項。

Ramzi Jammal 報告題目為「Regulating Innovative Nuclear Technologies」,首先介紹 CNSC 的使命、職權、組織、任務以及 2017 至 2018 年度之人力、財務及持照者申請與核發情形等。CNSC 是隸屬於加拿大政府的獨立核能安全管制機關,擁有 70 多年管制經驗,其使命為監管核能及核子材料的使用,以保護健康、安全及環境,並執行和平使用核能的國際承諾。

Jammal 主任也說明 CNSC 對於「小型模組化反應器」(Small Modular Reactor, SMR)的管制現況,大多數 SMR 設計概念仍基於既有核能電廠的技術與運轉經驗,但也採用許多創新技術。過去 CNSC 與其他許多國家的管制機關一樣,擁有審查傳統大型核能電廠新設計之經驗,面對 SMR 之設計審查及管制,CNSC 為能有效管制但不影響新技術發展,對於各類 SMR 設計,在符合加拿大的核安管制法規框架及管制要求下,允許申請者可提出靈活創新技術,以符合安全和環境保護的相關要求。在 SMR 設計審查方面,CNSC 也提供業者選擇「設計審查預審」(Pre-licensing vendor design review)進行方式,此進行方式非屬新核能電廠許可審查之必要程序,其目的在於驗證新核能電廠的設計及加拿大核能法規、標準及管制要求的可接受性,目前已有 10 個 SMR 設計申請預審。另外,也說明智慧眼鏡(smart glasses)應用於即時顯示輻射量測值及任務執行步驟的指引,及 Comanche Peak 核能電廠結合 Wi-Fi 網路與無線感測器做為遠端自動診斷系統等新技術,實際應用於核能電廠之使用及試驗情形。

(2)特別主題演講

Terrapower 副董事長兼智慧事業主任 Nathan Myhrvold 以「核能選項」為題進行

一個特別的大會主題演講，Myhrvold 強調福島的污染雖打擊核電的安全形象，但也促使核能界合作開發下一代技術，共同努力使核電更安全，更易於管理。未來全球電力需求將隨消費水平提升而增加，Terrapower 的目標是開發新型核反應器幫助滿足預期能源需求，新型核反應器可使用貧化鈾為核燃料並可無限期地維持分裂過程。Terrapower 正積極發展「行波反應器」(TWR)及「熔融氯鹽快中子反應器」(MCFR)，MCFR 在 2016 年 1 月獲得美國能源部以成本分攤方式提供五年總共 4 千萬美元的研發經費，TWR 則預備與中國核工業集團公司(China National Nuclear Corporation, CNNC)合作興建。

(3)產業觀點主題演講

接著是「產業觀點」(Industry Perspective)的主題演講，由加拿大 SNC Lavalin 全球核能業務執行副總 William Fox 擔任主席(表定主席全球核能業務總裁 Sandy Taylor 因故無法出席)，報告核能產業界發展現況與展望，講者包括中國核工業集團公司(CNNC)總工程師雷增光、韓國水力核能電力(Korean Hydro and Nuclear Power)執行副總 Sang-Wook Han、日本原子力產業協會(Japan Atomic Industry Forum)高級顧問 Takuya Hattori、以及 NuScale Power 營運長兼核能長 Dale Atkinson。

日本 JAIF 高級顧問 Takuya Hattori 報告題目為「Capacity Building is a Cornerstone for Sustainable Development of Nuclear Power」，Hattori 說明使用核能的優點及世界各國能源結構與核能發電所占比例，並說明核電可持續發展的條件，包含國家政策、核能安全、核安管制、技術創新、營運管理、國際交流、公眾信心及經濟競爭力等項。目前日本核電發展面臨人力資源的關鍵問題，分別有：(1)目前核電環境，無法吸引年輕人投入核工業；(2)沒有新的技術，可以吸引未來的研究發展；(3)有能力與經驗豐富的工程師，將陸續退休；(4)目前核電商業條件，無法維持高品質的核工業供應鏈；(3)因經費及未來前景發展未明，已停止研究新的反應器技術；(4)對於曾發生嚴重事故的反應器，欠缺除役相關技術；(5)一般民眾對輻射沒有正確觀念且易感到不安等。日本為健全核電人力資源的發展，由產官學組成 JNHRD (Japan Nuclear Human Resource Development Net)共同建置培訓核電人力，以滿足日本核電設施發展的需求，並解決前述人力資源問題。

(4)小組討論會

10 月 1 日下午有兩個「小組討論會」(Panel)，分別為：(1)由 2018 年太平洋盆地核能會議共同主席 Mimi Limbach 主持的聚焦於「核能科學溝通觀點」的「International Perspectives on Nuclear Energy Communications」。(2)由美國能源部核能高級顧問 Suzanne Jaworowski 主持的世界性小組討論會「The Millennial Nuclear Caucus Goes Global – Panel & Networking」。

2.專題研討論壇

本次會議的專題研討論壇共有 4 個主題，「核能的促進」(Advancing Nuclear Energy)與「核能的永續」(Sustaining Nuclear Energy) 分別有 2 個。

(1)核能的促進-前進的道路

10 月 2 日(週二)上午的專題研討論壇，由「核能的促進」專題研討論壇主席愛達荷國家實驗室「核能加速創新的通路」(Gateway for Accelerated Innovation in Nuclear)主任 Rita Baranwal 介紹與引言，第 1 個研討論壇「前進的道路」(The Path Forward)由美國核能產業委員會(Nuclear Industry Council)之總裁 David Blee 主持，討論「世界各地領先的核能研究、政策和產業組織是如何促進核能技術」，邀請講者包括澳洲核能科技組織(Australian Nuclear Science and Technology Organization, ANSTO)執行長 Adi Paterson、英國核能創新研究諮詢委員會(UK Nuclear Innovation Research Advisory Board)主席 Sue Ion、美國 GE-Hitachi 核能執行副總 Jon Ball、日本文部科學省(MEXT)原子能司司長 Takashi Kiyoura、加拿大核子實驗室能源計畫主任 Gina Strati。

澳洲 ANSTO 執行長 Adi Paterson 報告題目為「澳洲核燃料循環」(The Nuclear Fuel Cycle in Australia)，Paterson 說明 ANSTO 是澳洲核能專業技術的發源地，也是澳洲唯一有核反應器運轉營運的機構，並介紹澳洲 60 年來核能發展及亞太地區各國核能機組運轉、興建及未來機組規劃等情形；此外，澳洲鈾礦儲量豐富，持續開採鈾礦提供世界各國的核能電廠使用。

ANSTO 研究開發之開池式輕水型反應器(Open-pool Australian Light-water

Reactor, OPAL)於 2007 年 4 月開放使用，該反應器為一個多功能小型研究用反應器，主要用途為生產放射性同位素、輻射照射及中子束研究等。ANSTO 也推動核子科學的發展與技術研究，致力於建立核子科學對人類健康有益之核子醫學應用，以預防、診斷、治療及疾病檢測技術等研究；同時也利用核子技術及同位素技術，研究開發可解決澳洲及世界上一些具有挑戰性的環境問題(如水資源、環境變化及污染物影響等)；另外，並研究解決當前核反應器及未來新反應器之核燃料循環中的關鍵問題。ANSTO 也於 2017 年 9 月加入第四代反應器國際論壇(Generation IV International Forum, GIF)，與國際合作共同開發可行和性能更佳之新一代反應器。

日本文部科學省原子能司司長 Takashi Kiyoura 報告題目為「日本對於核能創新的未來展望」(Japanese Future Vision for Nuclear Innovation)，Kiyoura 說明福島事件後日本對於事故機組除役過程所做種種努力，包括除役路徑圖執行現況、公私部門協力分工情形，及日本境內核子設施的保留、除役規劃等。日本近年也因面臨核工系所學子、核電部門從業人員大幅減少的窘境，導致相關創新研究論文數量從原本僅落後美國的坐二望一之勢，變成已遠遠低於美、中兩國。既然研發創新能量不足，日本在有限資源下，所能做的僅有對內做好「能源選擇」與「資源整合」，以及對外強化國際合作一途。

英國核能創新研究諮詢委員會主席 Sue Ion 報告題目為「Advanced Modular Reactors」，Ion 首先說明建造傳統壓水式反應器機組之資本與財務成本，並分別就廠房結構、反應器、機械及電氣設備、儀控系統及營運管理等所占比例加以分析，相對於大型核能電廠興建所需時間較長與較高財務成本，開發預先完成模組化設計之組件較少、工期較短、資本成本較低的「小型模組化反應器」(SMR)似更為有利。

英國政府已啟動了一個新的兩階段式的「先進模組化反應器」(Advanced Modular Reactor, AMR)競賽計劃，第一階段將對 8 個 SMR 設計進行可行性研究，第二階段會選擇 3-4 家供應商加速其 SMR 的設計開發。

(2)核能的促進-技術的領導

第 2 個研討論壇「技術的領導」(Technology Leadership)由加拿大核子實驗室業

務發展副總 Corey McDaniel 主持，由領先的核能專家分享產業洞見、科學知識和促進核能技術之最新技術，邀請講者都是「小型模組化反應器設計」的關鍵技術參與者和供應商，包括「核能的未來」(Future of Nuclear Energy)作者美國愛達荷州國家實驗室 David Petti、英國國家核子實驗室(National Nuclear Laboratory)副主任 Gordon Bryan、上海應用物理研究所副所長戴志敏(未出席)、韓國原子能研究所(Korean Atomic Energy Research Institute) SMART 計畫主任 Han Ok Kang、加拿大 Terrestrial Energy 執行長 Simon Irish、美國 NuScale Power 營運及電廠服務副總 Ken Langdon。

NuScale Power 營運及電廠服務副總 Ken Langdon 報告題目為「NuScale Power-推進能源」(NuScale Power-Advancing Energy)。NuScale Power 是目前唯一的正接受美國能源部成本分攤資金 (5 年共 2.17 億美元)之 SMR 開發者，也是唯一的提出「設計認證」申請給美國核管會(NRC)之 SMR 開發者。NRC 已於 2018 年 4 月完成了其審查的第一階段(Preliminary Safety Evaluation Report and Requests for Additional Information)，預定於 2020 年 8 月完成其安全評估報告。「NuScale」模組化反應器是 SMR 發展競賽的領先者，是目前最受矚目的設計。

「NuScale」模組化反應器的原始設計係由能源部資助。NuScale Power 是在 2007 年從俄勒岡州立大學分離出來，在 NuScale Power 出現欠缺發展經費的問題後，Fluor Corporation 在 2011 年 10 月付出超過 3000 萬美元取得 NuScale Power 的 55%。

「NuScale」設計為 160 MWt/50 MWe 的一體化壓水式反應器，依靠重力使一次側冷卻水進行自然循環，因此並不需要使用冷卻水泵。「NuScale」將在工廠製造，使用直徑 3 公尺的壓力容器和對流冷卻，控制棒驅動裝置是唯一的移動組件。「NuScale」使用標準的壓水式燃料，鈾 235 濃縮度如同正常壓水式燃料束為小於 5% (長度僅為 2 公尺)，更換燃料週期為 24 個月。安裝於地面下一個充滿水的池內，直徑為 4.6 公尺、高度為 22 公尺的圓柱形圍阻體容器模組之重量為 650 噸，內含反應器和蒸汽產生器。一個標準的核能電廠將有 12 個模組，總容量大約 600 MWe。

(3)現有核能機組的永續－公用事業觀點

10 月 3 日(週三)上午的專題研討論壇，由「核能的永續」專題研討論壇主席加拿

大 SNC Lavalin Global Nuclear Business 執行副總 William Fox 介紹與引言，第 1 個論壇主題是「現有核能機組的永續－公用事業觀點」(Sustaining the Current Fleet – Utility Perspective)，由加拿大核子實驗室主任 Kathryn McCarthy 主持，討論「維續全球現有核能機組的關鍵因素」，邀請講者來自全球公用事業，包括美國 Arizona Public Service 公司電廠營運副總 Chuck Kharrl、韓國水力核能電力公司中央研究所(Central Research Institute)主任 Yunho Kim、美國 Southern Nuclear 公司創新與技術部門經理 Chris Comfort、日本電氣事業連合會(Federation of Electric Power Companies)核能部門總經理 Norio Atsumi。

韓國水力核能電力公司中央研究所主任 Yunho Kim 報告題目為「南韓的核能永續」(Sustainable Nuclear in Korea)，Kim 指出南韓因缺乏能源礦藏，政府視核能為國家能源戰略重要一環，並已長期積極發展核能，至今核電已占整體發電量 28.2%，是全球核電機組第 6 多的國家，而且技術能力領先，已連續多年創下功率損失和非計畫停機次數最少紀錄。但南韓內部反核聲浪高漲，民眾對於核電安全及核廢處理普遍存有疑慮。為此，南韓持續投入核電創新研發，寄望 2030 年問市的第四代反應器技術可以為這些問題提供技術解決方案。

由於環境和經濟原因，南韓現階段仍須維持核電，否則將增加燃煤生產以維持電價，同時也放棄過去發展核電的基礎。Kim 指出南韓核電的發展問題是社會問題，而非技術問題，為了維持南韓的核能永續性，發掘並解決現有核能電廠安全管理缺失，重新獲得民眾信任，是目前政府的當務之急。

(4)現有核能機組的永續－科技觀點

第 2 個論壇主題是「現有核能機組的永續－科技觀點」(Sustaining the Current Fleet – Science & Technology Perspective)，由前美國民用核能貿易諮詢委員會(U.S. Civil Nuclear Trade Advisory Committee)主席、Curtiss-Wright 核能部門市場發展副總 Gary Wolski 主持，討論「什麼關鍵的科學和技術支撐著全球現有核能機組的永續性？」，邀請來自全球核能供應鏈和研究組織的專家，包括美國電力研究院(Electrical Power Research Institute) Sherry Bernhoft、加拿大 CANDU 業主組織

(CANDU Owners Group)總裁 Fred Dermarkar、中國核動力研究設計院(Nuclear Power Institute of China)研究主任 Danrong Song、日本丸紅商事公共事業部(Marubeni Utility Services) Hirokazu Ofuji、愛達荷州國家實驗室美國輕水式反應器可持續性計畫(U.S. Light Water Reactor Sustainability Program)主任 Bruce Hallbert、加拿大核能產業組織(Organization of Canadian Nuclear Industries)主席 Ron Oberth。

CANDU 業主組織總裁 Fred Dermarkar 報告題目為「核能未來的持續挑戰與機會」(Challenges and Opportunities for Sustaining a Nuclear Future)，Dermarkar 表示，核能產業面對的主要挑戰包括資產管理、設備零件過時老化、網路安全、供應鏈安全、核廢料處理與核能電廠除役、核電營運與除役的成本與安全管制，但可把握機會推進核能的發展，包括核能人才的培育、資產價值的最大化、SMR 開發與興建、供應商間的合作及各核能組織間的協調合作。

美國愛達荷國家實驗室「輕水式反應器可持續性計畫」主任 Bruce Hallbert 報告題目為「輕水式反應器可持續性研究的科學與技術基礎」(Science and Technology Underpinning Light Water Reactor Sustainability Research)，Hallbert 說明此計畫目的為提高美國核能機組的安全與經濟效能，並延長這個可靠電力來源的運轉年限。目標包括：確保現有核能電廠的長期運轉，使用創新方法來提升輕水式反應器在近期和未來市場的經濟性與經濟競爭力，以及確保安全、增進可靠性、加強經濟性。研究重點領域包括材料、核能電廠現代化與風險資訊系統分析。

有關材料研究方面，可透過實驗、監測和材料驗證，來建立反應器壓力槽(RPV)脆化的預測模型；了解輻射促進應力腐蝕龜裂(IASCC)失效與應力腐蝕龜裂(SCC)的發生機制，預測並發展減緩策略；了解纜線劣化模式，預測性能並評估回復策略；建立纜線和混凝土結構的狀態監測技術；開發先進合金；開發先進的焊接技術，用於高輻射材料的焊接修復：焊接通常用於核部件的維修和升級，輻射照射材料的焊接可能導致脆化與開裂，為抑制裂隙發生，目前正在研究降低熱輸入焊接技術與減少焊接應力拉伸區的技術，以及使用類神經網絡監測焊接過程中的變數。

有關電廠現代化研究與發展方面，為解決現有電廠數位儀控技術的長期老化和可靠性問題以及提高電廠效率，採用作法包括：制定實施數位儀控技術長期現代化的策略；開發、測試與更新技術；減低技術、財務與監管風險；開發先進監測技術，以監測、檢測與減緩組件老化過程。在即時監測部分，採用作法包括：(1)以更具成本效益的「預測性維護」(Predictive Maintenance)策略替換現有「規定性維護」(Prescriptive Maintenance)策略，(2)透過核電公司和產業，此計劃正在開發風險告知的預測性維護策略，建立即時監測能力，使用資料驅動技術和先進感應器開發診斷和預測模型，建立風險評估能力，使用機率風險評估技術來提供決策信息，(3)透過電廠活動自動化來提高效率，以及降低運轉和維護成本。

加拿大核能產業組織主席 Ron Oberth 報告題目為「永續的核能-供應商的觀點」(Supplier Perspective)，Oberth 說明供應商在維持核能產業方面可發揮關鍵作用，透過創新、專注、合作與倡導來幫助降低成本、縮短時程、提高品質與安全與增進社會和政治層面的支持。在創新部分，將邀請中小企業供應商提出創新方案，以解決加拿大核能產業組織主要運營挑戰，並由供應商協助推動，項目主要包括機器人與無人機檢查作業、無紙化(電子化)、人工智慧、虛擬實境工具與 3D 列印技術的研發。

更換組件為老化核能電廠的一大挑戰，目前一些原本供應商已不再營業，部分組件已無法取得，透過商業級專用技術，可用於取得符合核能電廠需求的替換組件，節省成本和加快進度，並運用更多標準化作業流程，以降低成本。另外公營事業可鼓勵供應商結合多個合作夥伴提供綜合解決方案，透過共同合作開發創新產品與技術，並結合當地社區、地方政府與聯邦力量，創造就業機會並增進社會支持。

3.技術議程

本次會議發表的技術論文約有 140 篇，依論文性質分別於 39 個技術議程中發表，同一時間有 6-7 個技術議程同時進行，由與會人員視需要自行選擇參加。技術論文的主題共有下列 10 項：

安全與保安(Safety and Security)

運轉與維護(Operation and Maintenance)

新反應器的興建(New Build)

除役與廢料管理(Decommissioning Waste Management)

供應鏈與品質管理(Supply Chain and Quality Management Management)

燃料循環(Fuel Cycles)

進步型反應器(Advanced Reactors)

公眾諒解(Public Acceptance)

經濟(Economics)

醫藥 (Medicine)

4.晚宴及任務交接

大會在 10 月 3 日(週三)晚上舉辦晚宴及太平洋核能理事會主席交接，由下屆主席 Kamal Verma (加拿大 SNC Lavalin CANDU 6 Fleet Program 副總)、副主席王志(中國核學會副祕書長)致詞。下屆 PBNC (第 22 屆 PBNC)預定於 2020 年在墨西哥坎昆舉行，由主辦單位代表墨西哥核能協會會長 Javier Palacios 進行下屆 PBNC 之介紹。茲將歷屆 PBNC 開會時間與地點彙總如下表：

第 1 屆	1976	美國夏威夷	第 2 屆	1978	日本東京
第 3 屆	1981	墨西哥阿卡普爾科	第 4 屆	1983	美國舊金山
第 5 屆	1985	南韓漢城	第 6 屆	1987	中國大陸北京
第 7 屆	1990	美國聖地牙哥	第 8 屆	1992	台灣台北
第 9 屆	1994	澳洲雪梨	第 10 屆	1996	日本神戶
第 11 屆	1998	加拿大班芙	第 10 屆	2000	南韓漢城
第 13 屆	2002	中國大陸深圳	第 14 屆	2004	美國夏威夷
第 15 屆	2006	澳洲雪梨	第 16 屆	2006	日本青森
第 17 屆	2010	墨西哥坎昆	第 18 屆	2012	南韓釜山
第 19 屆	2014	加拿大溫哥華	第 20 屆	2016	中國大陸北京
第 21 屆	2018	美國舊金山	第 22 屆	2020	墨西哥坎昆

(三)發表論文

10 月 2 日下午於「燃料貯存選項的促進與挑戰的回顧」(Advances in Fuel Storage Options and Review of Challenges)技術分組會議發表「核二廠護箱裝載池安裝格架案之安全分析」(Safety Analyses for Installing Fuel Storage Racks in Cask Loading Pool at Kuosheng Nuclear Power Station)論文(如圖 1)，論文內容如附件一，簡報資料如附件二。

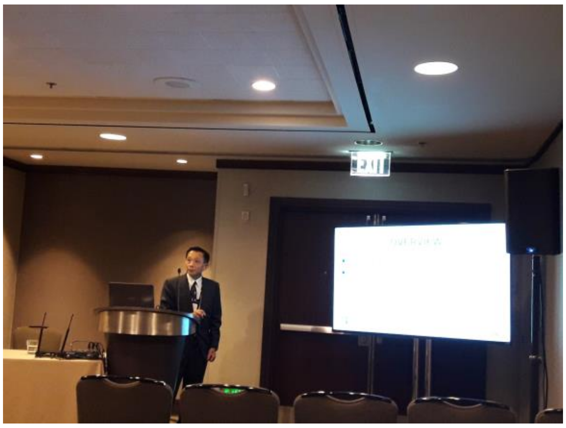

圖 1 發表論文

此論文重點在介紹本公司(台電)在「核二廠護箱裝載池安裝格架案之安全分析」的一些關鍵的安全議題與重要的經驗回饋，期能與國際核能社會分享國內的創新技術與重要經驗。核二廠一號機於 105 年 11 月大修時即面臨用過燃料池貯存容量不足，無法繼續運轉困境，本公司經探討各種可行性方案，採用護箱裝載池安裝 4 組龍門電廠二號機庫存格架的的創新作法。每部機組可增加 440 束用過燃料貯存空間，可讓機組繼續運轉 3 年。

本案依據原能會所發布審查規範之要求提出安全分析報告，安全分析報告內容包括 5 大技術範疇(臨界安全、熱流分析、結構分析、輻射安全、事故評估)之分析與評估結果，另針對相關議題亦提出詳盡的評估與說明。本案原能會審查極為嚴謹與詳盡，在參與人員共同努力下，本案於 106 年 4 月 6 日順利取得原能會核准。原能會於 5 月 19 日正式同意一號機之裝載池燃料貯存格架可啟用置放用過燃料，一號機在完成燃料挪移等作業後於 6 月 9 日併聯、6 月 16 日滿載發電，及時因應夏季尖峰用電需求。

本案關鍵的安全議題如：(1)熱流分析：護箱裝載池內池水溫度分布與局部沸騰；(2)結構分析：設計基準地震阻尼值及地震歷時時間長度之合理性與靈敏度分析；(3)輻射安全：裝載池北牆穿越管外側過高輻射劑量率之處理。(4)事故評估：使用「非耐單一失靈燃料廠房吊車」情況下格架墜落議題之評估處理。本案重要的經驗回饋如：(1)只貯存退出時間較長的老燃料有助於減輕不利影響，如裝載池內池水局部沸騰與穿越管外側過高輻射劑量率；(2)裝載池北牆穿越管外側輻射劑量率若超過限值，將依規定採取行政管制措施；(3)格架吊運程序有必要考慮吊車性能，先使用耐單一失靈之 150 美噸護箱吊車將格架吊入池內，再以燃料廠房 10 美噸吊車進行格架定位微調。

在綠能尚屬發展階段、夏季多次亮出「限電警戒」紅燈（備轉容量低於 90 萬瓩）的嚴峻情況下，本案順利推動與及時完成對台灣避免限電有關鍵性的貢獻。

(四)技術參訪

10 月 4 日(週四)由大會安排加利福尼亞大學柏克萊分校(簡稱柏克萊加大)核能技術中心的技術參訪活動。柏克萊加大位於舊金山灣區柏克萊市，是一所世界著名的公立研究型大學。柏克萊加大的校地總面積約為 5 平方公里，而主校區約 72 公頃。大學部約有學生 23,000 人，研究所約有 10,000 人。

柏克萊加大研究水平極高，截止 2018 年 9 月，共有 104 位教職員或校友為諾貝爾獎得主、位列世界第三。「原子彈之父」羅伯特·歐本海默、「氫彈之父」泰勒均曾長期擔任柏克萊加大教授；物理學家恩尼斯特·勞倫斯教授在此發明了迴旋加速器，藉由迴旋加速器柏克萊加大以及勞倫斯柏克萊國家實驗室的研究人員共發現了 16 種化學元素，位居世界第一，其中鉳(Berkelium，原子序為 97)和鉲(Californium，原子序為 98)更以柏克萊加大的名字來命名，而鐒(Lawrencium，原子序為 103)和𨭎(Seaborgium，原子序為 106)則是以此校的勞倫斯和葛蘭·希柏格(Glenn T. Seaborg)的名字來命名的。二次大戰時期，柏克萊加大的勞倫斯放射實驗室承包了美國軍方的原子彈研發計劃。1942 年，羅伯特·歐本海默教授被任命領導曼哈頓計劃的科學部門。

此參訪活動巡覽校園周圍的歷史地點，包括葛蘭·西柏格在 1941 年 2 月 23 日發現鈽的 Gilman Hall，基此，該建築物被選定為國家歷史地標，校園內的薩瑟塔(Sather Tower)是加州大學最顯著的標誌，常被稱為「鐘樓」(The Campanile)，完成於 1914 年，1917 年首次向公眾開放。這座塔高 307 英尺(93.6 米)，是世界第三高的鐘樓。鐘樓旁有一座南大樓(South Hall)，是校園內最古老的建築物。

柏克萊加大核能技術中心的技術參訪活動由核子工程系系主任 Peter Hosemann 主持，先進行系務相關(學術、研究計畫及教授群)介紹，核子工程系成立於 1958 年，目前該系大約有 100 名研究生與 100 名大學部學生，共有 8.5 名約當全職(FTE)教授，前一會計年度(2016/2017)研究經費為 1.17 千萬美元。

接著參觀重要實驗室，包括：

(1)核子材料測試(nuclear materials testing)實驗室:目前的研究重點包含運用離子束照射加速材料測試、反應器組件的結構材料腐蝕等。

(2)熱流(thermal hydraulics)實驗室：備有廣泛的實驗能力，可進行縮小尺寸試驗以

驗證先進反應器包括氟鹽冷卻高溫反應器(FHR)的熱流程式。熱流實驗室廣泛依賴「LabView」平臺進行資料獲取和控制，目前的實驗設施包括「小型整體效應測試」(Compact Integral Effects Test)，如圖 2 所示，可複製使用被動安全系統之 FHR 的穩態和暫態熱輸送;目前該實驗室也為核安監管機構審反應器執照申請提供實驗驗證數據。

圖 2 小型整體效應測試設施

(3)高通率中子產生器(High Flux Neutron Generator)：是一個基於雙離子源的 DD 中子產生器(氘與氘反應產生中子)，如圖 3 所示。

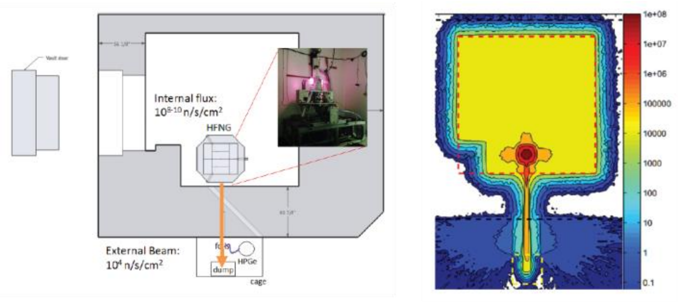

Figure 1 Figure 2

圖 3 高通率中子產生器

(4)輻射檢測和成像(radiation detection and imaging)。

(5)程式開發與計算(computer-code development and computation)。

這些實驗室對廣泛的應用進行研究，包括進步型反應器安全和執照申請、現有和進步型反應器的材料可靠性、檢測和成像醫學、以及核能安全。

三、心得與建議

1. 「小型模組化反應器」(SMR)為本次會議專題研討論壇的焦點主題，大會邀請美國、英國、加拿大、韓國及中國大陸的的關鍵技術參與者和供應商，探討 SMR 之發展。

   「小型模組化反應器」之定義為 300 MWe 以下，在工廠製造，完全建好後再送到廠區安裝之反應器設計。雖然 SMR 比傳統反應器產生較少的電力，但可以更靈活地使用和更有效地運作。SMR 通常具有被動的安全功能和固有的安全功能，可以在沒有外部電源情況繼續安全運作，即可以有效地處理 2011 年日本福島事故在海嘯來襲之際所面對的問題。

   美國、英國及加拿大最近幾年在 SMR 的發展採取一致的做法，都是企圖以設計競賽評審，找出最佳的 SMR 設計，然後藉由政府的補助經費推動 SMR 技術建立，以促成 SMR 的興建。發展 SMR 的動機也相同，都是要減少碳排放，以及建立 SMR 的製造能力、成為創新核能技術的領導者。美國、英國及加拿大都是民用核能的前鋒，有嚴謹的核能安全管制，以下為這三個國家的 SMR 發展概況。

   美國的發展概況：

   美國起步最早，SMR 在美國被認為是可在逐步受侵蝕的核能技術市場重新取得領導地位的機會，能源部推動 SMR 的計畫在 2012 年 1 月正式起步，能源部藉由成本分攤的方式推動 SMR，政府發展 SMR 的補助經費，估計五年總共為 4.52 億美元。在美國有四個發展中的主要輕水式 SMR 設計：(1) Babcock & Wilcox 的 180 MWe「mPower」；(2)西屋的 225 MWe「WSMR」；(3) NuScale Power 的 50 MWe「NuScale」；(4) Holtec International 的 160 MWe「HI-SMUR」。所有四個設計都申請能源部資助，其中「mPower」及「NuScale」設計獲得源部同意援助。在能源部與 SMR 核電公司簽署的協定裡，明確規定所設計的反應器必須在國內生產，以增強美國的製造能力和提高出口能力。按照能源部的規劃，SMR 將在 2025 年左右商業化，屆時美國將成為全球 SMR 行業的領導者。

   SMR 具有被動的安全功能和固有的安全功能，在福島事故後，SMR 訴求的簡單化和安全性似已更有賣點。但現實上 SMR 的發展仍有很大難度，由於缺少投資者與顧客意願，原本領先者 Babcock & Wilcox 已在 2016 年減緩設計發展工作。NuScale Power

基於積極推動國內外合作計畫，已取代原本領先者 Babcock & Wilcox，成為目前 SMR 發展的領先者。NuScale Power 的 SMR 正在接受美國 NRC 的「設計認證」審查，是第一個也是迄今為止唯一的 SMR。「猶他州市電力系統協會」(Utah Associated Municipal Power Systems)正計畫在能源部的愛達荷州國家實驗室的一個地點開發一個 12 個模組的 NuScale 電廠，預期在 2020 年中期部署。

另外，能源部在 2015 年建立「核能加速創新的通路」(GAIN)倡議，GAIN 為整個能源部及其國家實驗室範圍廣泛的功能提供取得協助的單一窗口。GAIN 在 2016 年 1 月提供最多達 4 千萬美元的經費給 X-energy 以發展其「Xe-100 pebble-bed HTR」的設計，和最多達 4 千萬美元給 Southern Co 以發展其「熔融氯鹽快中子反應器」(MCFR)，MCFR 係由 Southern Co 與 TerraPower 和橡樹嶺國家實驗室共同發展。能源部另在 2018 年 4 月選擇了 13 個專案，共提供 6000 萬美元的先進核能技術成本分攤研發資金。

英國的發展概況：

在英國方面，英國發展 SMR 的原因與美國類似，英國需要核能協助以達成法定的碳排放目標，並替換老舊電廠。英國是第 1 個通過立法規定減碳目標的國家，在面臨未來十年有大量電廠除役之際，核電被英國政府視為電力部門重要的減碳與穩定電力供應選項，因此近年來提出許多政策措施來刺激國內核電興建的投資。

有鑑於歐美大型核能電廠建廠成本不斷飛漲、工期延宕及充滿技術和政治風險，SMR 被視為核能工業問題的解決方案。英國政府在 2015 年 11 月宣佈在未來的五年要投入 2 億 5000 萬英鎊做核能研究發展，以「重振英國的核能專業能力，成為全球領先的創新核能技術的國家」，主要部分就是拿來辦競賽，為英國找出最佳價值的 SMR 設計。2016 年 3 月 17 日英國「能源與氣候變化部」(Department of Energy and Climate Change, DECC)正式發起 SMR 設計的第一階段競賽。在 2017 年 12 月，DECC 的後繼部會「商業、能源及產業策略部」(Department for Business, Energy and Industrial Strategy, BEIS)宣佈競賽結束，英國政府啟動了一個新的兩階段式的「先進模組化反應器」(Advanced Modular Reactor, AMR)競賽計劃，目標在整合更廣泛的反應器類型。第一階段對 8 個 SMR 設計進行技術和商業可行性研究，由下列 8 個組織獲得可行性研究的合約：Advanced Reactor Concepts LLC、DBD Limited、Blykalla Reaktorer Stockholm AB (LeadCold)、 Moltex Energy Limited、Tokamak Energy Ltd、U-Battery Developments Ltd、Ultra Safe Nuclear Corporation、Westinghouse Electric Company UK。

第二階段會選擇 3-4 家供應商加速其 SMR 的設計開發，總資助金額高達 4400 萬英鎊。除此之外，英國政府和法國電力公司、西屋公司、NuScale 以及勞斯萊斯等製造商在 SMR 方面都進行了密切的合作與開發活動。

加拿大的發展概況：

加拿大起步較晚，但企圖心極強，想要後來居上。2017 年年初，加拿大核子實驗室(Canadian Nuclear Laboratories, CNL)發表了一項長期戰略，其目標是在 2026 年前在 Chalk River 地區選址建置新的 SMR。CNL 並在 2017 年 6 月發布一項邀請(Request for Expressions of Interest)，希望 SMR 技術開發者、潛在用戶、以及利益相關者(例如：潛在場址社區、核能供應鏈和學術研究機構)，針對在加拿大將 SMR 技術推向市場提出意見，讓 CNL 更瞭解其現有能力、技術差距、需求、及整體市場利益。共有 80 個機構向 CNL 提出意見，其中有 19 個機構表示有意願建構 CNL SMR 的示範原型。

CNL 後續在 2018 年 5 月發布一項 SMR 示範專案的邀請。SMR 示範專案包括四個不同階段：第一階段是可自由選擇的的「資格預審」(Pre-Qualification)，將根據初步標準對申請者進行審查。第二階段是「盡職調查」(Due Diligence)，申請者必須滿足更為嚴格的財務要求，因此將對申請者的資金情況和專案成本進行全面評估。第三階段是「土地安排和其他合約的談判」(Negotiation of Land Arrangement and Other Contracts)，最終將與場址的擁有者加拿大原子能有限公司 (AECL) 簽署一項場址處置協定。 第四階段是「專案執行」(Project Execution)，將包括 SMR 機組的執照申請與施工、測試、啟用、運轉和最終除役。

CNL 在 2018 年 6 月宣佈收到第一期 4 個 SMR 示範專案的申請，包括 3 個第一階段及 1 個第二階段的申請者，並預期未來會有更多申請者。CNL 希望未來能夠成為 SMR 技術開發的全球中樞。

CNL 在加拿大安大略省(Ontario)的 Chalk River 實驗室是 National Research Universal (NRU)反應器的所在地，經過 60 年的運轉，NRU 已於 2018 年 3 月 31 日關閉；在此期間，NRU 已成為世界上規模最大，功能最全面的高通率研究用反應器

之一，同時也是醫療用放射性同位素的重要供應者。配合 NRU 的關閉，Chalk River 地區是加拿大建構 SMR 示範原型最優先的規劃場址。

2. 中國大陸是目前興建中核能機組最多的國家，大陸於 2018 年 10 月 31 日有 45 部核能機組運轉中，12 部機組興建中，2017 年核能發電比例為 3.94%的電力。

大陸積極發展核電技術，2007 年從西屋電氣引進「AP1000」，2011 年在「消化吸收」AP1000 的基礎上完成「AP1400」的初步設計，2013 年研發出用於出口的「華龍一號」。華龍一號是中國廣核集團(CGN，簡稱中廣核)的 ACPR1000 與中國核工業集團公司的 ACP1000 合併融合的產物。2016 年中廣核與法國電力集團拿下了英國 Hinkley Point C 新建核電計畫，另中廣核董事長賀禹於近期表示已有十多個國家對華龍一號表現出濃厚興趣。中廣核的積極行銷核電已導致美國政府的壓抑行動。

以往中廣核人員常受邀出席太平洋盆地核能會議報告中國核電的發展，2018 年會議並沒有中廣核人員出席，可能是感受到關係緊張，不敢惹麻煩。似乎有點巧合的是，美國政府於 10 月 11 日發出限制令，限制向中國出口核能技術，內容包括禁止出口核產品及技術到中國最大的核電公司中廣核，至於能否對中廣核以外的中國公司出口核產品或技術，美國官員將根據具體情況逐一審查。

本次會議中國核工業集團公司總工程師雷增光獲邀於「核能的永續與促進 – 產業觀點」(Advancing and Sustaining Nuclear Energy – Industry Perspective) 專題研討論壇報告「中國核電的發展現狀與展望」(The Development Status and Prospect of China’s Nuclear Energy)，要點如下：

中國大陸核電發展分為三個階段：初始階段(1980 ~ 1995)、適度發展階段(1995-2005)、擴大發展階段(2005 迄今)。達到的成果包括：(1)一個開發和利用鈾資源的完整核燃料循環系統; (2)一個完整的核電工業系統; (3)一群運轉良好、維持良好安全記錄的高品質核能電廠; (4)為核能研究、開發、建造、運營和維護，培養了一個優秀的人力資源團隊。

中國大陸在巴黎協定中表示，到 2030 年，中國將把其每單位 GDP 的二氧化碳排放量從 2005 水準降低 60%至 65%。核能將在實現這一目標方面發揮關鍵作用。核能是優化中國能源組合、促進中國生態環境的重要解決方案。雖然核能的發展受到福島事故的影響，但總體趨勢正在朝著「更安全更環保」的目標發展。

3. 核能的溝通宣導與公眾接受是核能議題成敗的關鍵要素，公眾要的不只是「安全」，還要能「安心」，一再強調安全反而可能讓人不安，本次會議「核能溝通觀點」專題研討論壇提出的一個重要新觀點是：「信使比信息重要」(Messenger is more important than Messenge)，「信賴的人選」加上「有感的論點」是達到有效溝通的關鍵。

4. 綜觀各國核能產業主要面對的挑戰，包括:資產管理、設備零件過時老化、網路安全、供應鏈安全、核廢料處理與核能電廠除役，關係到核電營運、除役與核廢料處理的成本與安全管制等，上述各項亦為我國核能發展面臨的問題，值得持續汲取相關經驗，做為我國核能產業之參考借鏡。

附件一

SAFETY ANALYSES FOR INSTALLING FUEL STORAGE RACKS IN CASK LOADING POOL AT KUOSHENG NUCLEAR POWER STATION

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Abstract- *Even with the two Spent Fuel Pool (SFP) reracking projects to expand the capacity to 4398 cells, Kuosheng Unit 1 had practically no unfilled SFP cells such that operation was prohibited from November 30, 2016. After a careful evaluation of the possible measures to restore refueling ability, the Taiwan Power Company has elected to install four 11x10 storage racks within the Cask Loading Pool. The additional 440 cells could allow operation for two more fuel cycles, about 3 more years. All the aspects (criticality, thermal-hydraulic, structure, radiological, and accident) specified in the Republic of China Atomic Energy Council's review guidance are evaluated. Additionally, distinctive subjects such as transportation, installation, post-Fukushima safety enhancement measures, and emergency/recovery plans are thoroughly addressed. The Safety Analysis Report (SAR) was submitted in August 2016. As the safety issues were thoroughly examined, many unique and significant issues worth exploring have been identified and additional analyses requested. After all the issues resolved, the SAR was approved in April 2017. For Unit 1, rack installation was completed and the unit was back online in June 2017. The timely and successful implementation of this contingency project is extremely critical for achieving a stable power supply in Taiwan.*

I. INTRODUCTION

The Kuosheng Nuclear Power Station consists of two BWR-6 units currently rated at 3001 MWt (~1020 MWe). Each unit has a Spent Fuel Pool (SFP) for storage of spent fuel. An additional storage capacity of 663 assemblies exists in the Upper Fuel Pool (UFP), however, the UFP is not allowed for storage of spent fuel when the reactor is critical. The second SFP reracking project (reported in 15th PBNC) completed in 2004 adds six new rack modules to the SFP, and the total cell count is increased to 4398 storage cells.[1] With the expanded capacity, however, the SFP for Kuosheng Unit 1 had practically no unfilled cells for discharged fuel such that further operation was prohibited from November 30, 2016 due to inability to refuel. The Taiwan Power Company (TPC) had originally planned to move some spent fuel assemblies to a dry storage facility using NAC Magnastore technology (27 casks each holding 87 fuel assemblies). The AEC issued a license for the Kuosheng dry storage facility in August 2015, however, final approval for the facility is blocked by the local government.

After a careful evaluation of the possible measures to restore refueling ability (mainly using the UFP for long-term storage of old spent fuels, or sending spent fuel abroad for reprocessing), TPC has elected to expand the spent fuel storage capacity by installing four 11 by 10 storage racks (from the inventory of Lungmen Unit 2) in the Cask Loading Pool (CLP). The additional 440 cells could allow operation for two more fuel cycles, about 3 more years. As a prudent measure to minimize the adverse impacts such as localized boiling in the CLP and excessive exposure rates, the spent fuels allowed to be stored are limited to old spent fuel assemblies offloaded from the first to the forth fuel cycles (decayed for more than 29 years). The Pacific Engineers & Constructors Ltd. (PECL) was awarded this contract.

The effects on all aspects (criticality, thermal-hydraulic, structure, radiological, and accident) specified in the Republic of China (ROC) Atomic Energy Council's (AEC's) review guidance are evaluated,[2] and all the acceptance criteria are met. In addition, distinctive subjects such as rack transportation, installation, emergency plans, recovery plans, neutron absorber panel test plans, applicability of the post-Fukushima safety enhancement measures, and the integrity of the CLP are thoroughly addressed.

The Safety Analysis Report (SAR) for this project was submitted to the AEC on August 18, 2016. A review task force was organized by the AEC to perform a very strict review. As the safety issues were thoroughly examined during the review process, many important issues have been identified and additional analyses requested, such as: (1) local temperature distribution analysis to demonstrate that localized boiling will not occur in the CLP, (2) sensitivity analyses for the total duration of time history and damping value utilized in the rack dynamic analyses, and (3) safety margin with respect to the fuel damage for the rack dynamic analyses.

These issues required very extensive efforts to resolve. With the cooperative efforts by the TPC team, however, all the issues were fully clarified and SAR was approved by the AEC on April 6, 2017. For Kuosheng Unit 1, the rack installation work was completed in May, and the unit was back online on June 9, 2017.

II. GENERAL DESCRIPTION

II.A. Pools and Racks

The SFP consists of two discrete pools, and the CLP is adjacent to the SFP, as shown in Fig. 1. The second SFP reracking project only adds six new rack modules supplied by ENSA to the SFP. Except a specialty rack, all the existing rack modules supplied by Holtec from the first SFP reracking project continue to be used. The total cell count is 4398 storage cells.

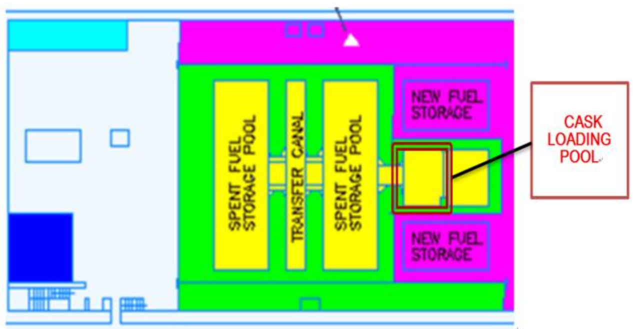

Fig. 1. Spent Fuel Pool and Cask Loading Pool.

The Cask Loading Pool after the installation of the storage racks is shown in Fig. 2.

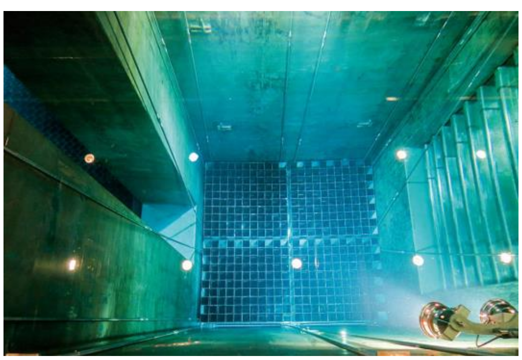

Fig. 2. Cask Loading Pool after installation of the racks.

The Lungmen racks are also supplied by ENSA, and the main material are the same as the existing racks. The storage rack cell consists of a squared stainless steel tube surrounded by four fixed Boral sheets. The minimum Boral panel B-10 areal density (0.020 g/cm^2) is the same as the existing ENSA racks, and is much higher than that of the existing Holtec racks (0.012 g/cm^2). The cell pitch is 167 mm, which is substantially larger than the 159.5 mm for the existing racks. As part of the installation process, the various pipings (makeup and drain pipes) and other protrusions on the pool wall are removed. Also, the existing stiffening frame around the Lungmen racks is supplemented with a displacement restraint to provide horizontal support.

II.B. Safety Analysis Report

The SAR documents the design and analyses performed to demonstrate that this project meets all requirements, in particular, the AEC's review guidance for the SFP reracking. The scope of evaluations for the SAR includes the five technical aspects specified in the review guidance:[2]

1. Criticality
2. Thermal-Hydraulic
3. Structure (Mechanical, Material and Structural)
4. Radiological
5. Accident

Additionally, distinctive subjects such as rack transportation, installation, emergency plans, recovery plans, neutron absorber panel test plans, applicability of the post-Fukushima safety enhancement measures, and the integrity of the CLP are thoroughly addressed.

II.C. Licensing Activities

The SAR for this project was formally submitted on August 18, 2016. The AEC review process was separated into two stages: (1) the acceptance review, and (2) the detailed technical review. For the acceptance review, 21 questions were raised by the AEC on the completeness of the SAR. After all the questions clarified and the SAR revised consequently, the SAR was accepted for the detailed technical review on September 20, 2016.

For the detailed technical review, a review task force consists of scholars and experts of various aspects (including 11 external members and 13 AEC members) was organized by the AEC to perform a very thorough review. The task force was separated into three functional groups:

A. Accident/Criticality/Thermal-Hydraulic
B. Structure/Material
C. Radiological/Waste Treatment

A joint review meeting for all the three groups was conducted on September 21, 2016, and subsequently 10 review meetings for the individual group were conducted. Four rounds of Request for Additional Information (with 62/85/40 questions for Groups A/B/C, respectively) were raised by the AEC. Responses to those questions were prepared jointly by the TPC and PECL. Some of the review questions required rather extensive efforts to address adequately, and follow-up questions to the initial questions of each round were raised if the issues were not considered fully clarified. With the cooperative efforts by the TPC team, all the issues were fully clarified and SAR was approved by the AEC on April 6, 2017.

III. SAFETY ANALYSES AND KEY ISSUES

III.A. Criticality Analysis

III.A.1. Methodology and Results

The criticality analyses were performed with the multigroup transport theory code CASMO-4, and three-dimensional Monte Carlo code MCNP5. The CASMO-4 code was used as the primary method as well as a means of evaluating small reactivity increases associated with manufacturing tolerances. The MCNP5 code was used for independent verification of the CASMO-4 results, and when needed, to evaluate accident conditions.

The criticality analyses were performed using conservative methodology similar to that of the previous reracking projects. The analyzes were performed for the two fuel types: GE8x8-2 and ANF8x8-2. CASMO-4 calculations were performed for all the lattices of the two fuel types. The effect of abnormal and accident conditions are evaluated for: (1) temperature and water density effect, (2) Zircaloy fuel channel distortion, (3) abnormal location of a fuel assembly, (4) dropped fuel assembly, and (5) fuel rack lateral movement. The maximum k_{eff} (including all uncertainties) of the 9 lattices is 0.80768, which is far less than the 0.95 limit. The maximum effect of abnormal and accident conditions are conservatively evaluated to be 0.00355 (for abnormal location of a fuel assembly).

III.A.2. Key Safety Issues/Additional Analyses

Since the Boral panel B-10 areal density and the cell pitch for the Lungmen racks are more conservative with respect to criticality safety than those for the existing racks, there is no real safety concern. Rather, the TPC was requested to perform CASMO-4 calculations for the racks from Lungmen, the first reracking and second SFP reracking, to demonstrate the criticality safety margin. The results demonstrate that infinite multiplication factor for the Lungmen racks is indeed lower than those for the first reracking and second SFP reracking.

III.B. Thermal-Hydraulic Analysis

III.B.1. Methodology and Results

The thermal-hydraulic analysis evaluates the following items: (1) maximum pool temperatures, (2) maximum local water and cladding temperatures, and (3) the time available for corrective actions after loss of forced cooling. Since the decay heat load for fuels discharged more than 29 years is very low, the increase in the total heat load is quite low (~2%) and the effects are insignificant. The calculated maximum bulk temperatures remain below the acceptance criteria, and the maximum local water temperature in the SFP is lower than the saturation temperature so that localized boiling will not occur. With loss of forced cooling, the maximum boiloff rate is only ~36 gpm, which is significantly less than capacity of makeup water sources (50 gpm). The time-to-boil is greater than 11 hours and time for water to drop to a level of 3 meters above top of fuel bundles is greater than 87 hours, thus, a sufficient time is available for corrective actions to align and supply sufficient water, from a variety of sources, to the SFP.

III.B.2. Key Safety Issues/Additional Analyses

1. CLP water temperature distribution

There is no forced cooling flow injected into the CLP, and the decay heat load in the CLP is removed by natural convection of CLP water to the SFP through a gate (Gate 3) connecting the two pools. However, the bottom of Gate 3 is approximately 17 feet above the bottom of the CLP, and it was suspected that this might impede the cooling of the CLP, resulting in excessive local water temperature. To clarify this issue, a fluid dynamics and temperature distribution analysis was requested by the AEC.

The GOTHIC code was used to perform the analysis for storage of old spent fuel assemblies with very low decay heat. Figure 3 shows that the maximum variation in the CLP water temperature is only ~0.7℃, thus, Gate 3 could allow sufficient natural convection of the CLP water to the SFP, and localized boiling will not occur. However, excessive local water temperature could be a concern if newly discharged fuel assemblies are stored in the CLP.

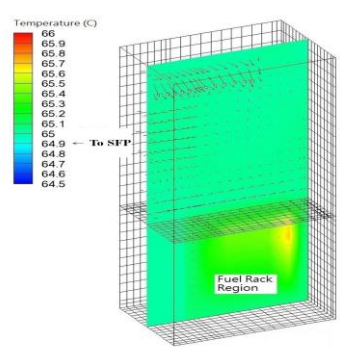

Fig. 3. Cask Loading Pool water temperature distribution.

2. Abnormal Refueling with Full Core Discharged to UFP

The TPC was requested to calculate the maximum pool water temperature for “abnormal refueling case with full core discharged to UFP”.

It is stated in the Kuosheng FSAR that "During a full core outage when elements are stored in the upper pool, those **elements that have decayed the longest in the spent fuel pools may be moved into the upper pool storage racks**. This operation is done to reduce the heat load in the upper pool ...". Thus, "abnormal refueling case with full core discharged to UFP" is not considered an applicable case.

Nonetheless, the maximum UFP water temperature for abnormal refueling case with the highest UFP water temperature was estimated using the relationship that the temperature rise (pool temperature minus heat exchanger inlet coolant temperature) is proportional to the heat load, and is inversely proportional to the coolant flow rate. The relationship has been validated using the results from the second SFP reracking. The maximum UFP temperature from the second SFP reracking is 72.53℃, while the limit value is 100℃. With the increased heat load, the maximum UFP temperature estimated using the relationship is 74.3℃, still far below the limit value.

III.C. Structure Analysis

III.C.1. Methodology and Results

The structure analyses include two main parts: (1) rack dynamic analyses and structural evaluation, and (2) fuel building and CLP structural integrity evaluation.

The structure analyses are complicated by the new finding for the nearby Shanchiao fault. In the Kuosheng FSAR, the design basis Safe Shutdown Earthquake (SSE) is anchored at a peak ground acceleration (PGA) of 0.4g in the horizontal direction, and the Operation Basis Earthquake (OBE) is one half of the SSE. As required by the AEC, the effects of the new finding are considered using assumptions provided in Table I: For the newly added systems and components such as the racks, the effects are accounted for by the use of a 0.67g Design Basis Earthquake (DBE). For the existing structure, DBE and OBE in the FSAR are continued to be used, and the Seismic Margin Assessment (SMA) methodology is used to address the Beyond Design Basis Earthquake (BDBE) using a PGA of 0.67g.

TABLE I. Acceleration assumptions for the newly added systems and the existing structures

	OBE	DBE	BDBE
Newly Added Racks	0.2g FSAR	Envelop 0.67g SSE & 0.67g CR-0098	-
Existing Structure CLP Structure	0.2g FSAR	0.4g FSAR	0.67g SSE
Existing Structure Fuel Building	0.2g FSAR	0.4g FSAR	0.67g SSE

1. Rack Dynamic Analyses and Structural Evaluation

The analyses are performed in compliance with AEC's review guidance,[2] and the USNRC's Standard Review Plan (SRP). A non-linear dynamic analysis of the fuel storage racks was performed using ANSYS to obtain the forces, displacements, deformations, etc. The analyses were based on the simulation of the DBE and OBE in accordance with SRP 3.7.1, "Seismic Design Parameters". The synthetic time-histories in three orthogonal directions (N-S, E-W, and vertical) are generated in accordance with SRP 3.7.1. The applicable loads and their combinations are consistent with SRP 3.8.4. Stress calculations of the racks were performed to demonstrate the compliance with the requirements of the ASME Boiler and Pressure Vessel Code. All the calculated stresses meet the requirements.

2. Fuel Building and CLP Structural Integrity Evaluation

Design Basis Seismic Evaluation for Fuel Building was performed due to the installation of the racks in the CLP. The original FSAR loading conditions and acceptance criteria are adopted. The evaluation results show that the seismic impact of the additional fuel bundles and racks in the CLP is very limited. For the most limiting load combination, the minimum safety factor remained above 1.0. All major structural components still meet the design basis seismic requirements.

In addition, a SMA High-Confidence of Low Probability of Failure (HCLPF) analysis was also performed for the Fuel Building. The HCLPF capacity is equal to the earthquake magnitude at which the strength limit is reached. The methodologies and criteria of EPRI NP-6041-SL were applied for the HCLPF calculation. The evaluation results for major structural components show that the SMA criteria are met.

3. Spent Fuel Pool Integrity Evaluation

To address Fukushima Near-Term Task Force Recommendation 2.1, a "SFP/CLP Integrity Evaluation" was performed in accordance with EPRI-1025287, "Seismic Evaluation Guidance". The results show that the uncovering of the fuel bundles during normal operation and refueling outage will not occur within 72 hours.

III.C.2. Key Safety Issues/Additional Analyses

1. Total duration of time history

The synthetic time-histories utilized in the rack dynamic analyses are generated in accordance with the provisions of SRP 3.7.1. The total duration of time history used is 24 second, which meets the SRP 3.7.1 requirement of at least 20 seconds. Since the 24 second duration is shorter than that experienced previously in the major earthquakes in Taiwan, a sensitivity analysis with a longer duration was requested. A sensitivity analysis with a 48 second duration was performed for the most limiting case. The calculated maximum stresses for the sensitivity case are bounded by the base case with a 24 second duration, since the maximum amplitude of acceleration is somewhat lower for the case with a 48 second duration.

2. Damping values

The structural damping values of 2 percent for OBE and 4 percent for SSE assumed in the rack dynamic analyses are consistent with Regulatory Guide 1.61. However, the 4 percent damping for SSE is not consistent with the 3 percent value specified in the Kuosheng FSAR for welded structural assemblies for SSE. TPC was requested to provide justification, as summarized below: It is stated in Section IV (3) of the USNRC OT Position that "For plants where dynamic input data such as floor response spectra or ground response spectra are not available, necessary dynamic analyses may be performed using the criteria described in Section 3.7 of the Standard Review Plan. The **ground response spectra and damping values should correspond to Regulatory Guide 1.60 and 1.61** respectively…".[3] The rack dynamic analyses for this project were based on the simulation of the 0.67g SSE in accordance with SRP 3.7.1 guidance. The floor response spectra were generated considered the new soil related input parameters, soil-structure interaction effects as well as the increased live load due to fuel storage, and a peer review has been completed by the National Center for Research on Earthquake Engineering. Thus, the rack dynamic analyses meet the requirements of the AEC's review guidance and USNRC OT Position, and the use of 4 percent damping for SSE is considered adequate. For completeness, a sensitivity analysis was performed for the most limiting case with a 3 percent damping for SSE. The calculated maximum stresses for the 3 percent damping are slightly larger than those for the 4 percent damping. The calculated maximum primary stresses in the rack module structure for the 3 percent damping continue to meet the ASME requirements.

3. Safety margin with respect to the fuel damage

The permissible lateral load on a spent fuel assembly has been studied by the Lawrence Livermore National Laboratory (LLNL). The LLNL report UCID-21246 "Dynamic Impact Effects on Spent Fuel Assemblies" states that "...for the most vulnerable fuel assembly, axial buckling varies from 82g's at initial storage to 95g's after 20 years' storage. **In a side drop, no yielding is expected below 63g's at initial storage** to 74g's after 20 years' storage", thus, the acceptance criterion adopted by the USNRC for the maximum lateral acceleration is 63 g.

The estimated maximum impact load corresponds to a deceleration of about 8.15 g, and the nominal factor of safety against fuel failure is approximately 7.7.

III.D. Radiological Evaluation

III.D.1. Methodology and Results

The radiological aspects of the increased thermal and radiological releases from the facility under normal as well as accident conditions are addressed. The exposure rates from the SFP water, the occupational exposures, the generation of radioactive waste, and accident conditions (including fuel handling accident, rack or cask drop accident, offsite doses for loss of cooling to the SFP) were evaluated. One major issue has been identified: the calculated exposure rate at the outer surface of the North wall for the CLP (Point 1 in Fig. 4), 0.0514 mSv/hr, exceeds the 0.05 mSv/hr limit for "Radiation Area", due to the 12-inch piping penetrating through the wall. To address this issue, any area with a measured radiation level in excess of 0.05 mSv/hr will be classified and posted as "Radiation Area" and administrative controls will be implemented following the rack installation.

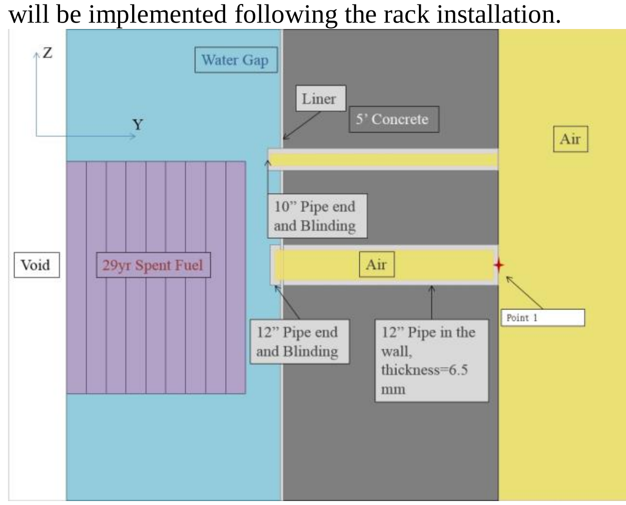

Fig. 4. Radiological evaluation model for the north wall.

III.D.2. Key Safety Issues/Additional Analyses

The exposure rates from the SFP water, the occupational exposures, and offsite doses for loss of cooling to the SFP were originally evaluated with SFP radionuclide concentrations from a single measurement. TPC was requested to address of the effect of concentrations variation during a fuel cycle. The relevant calculations were re-performed with maximum concentrations for the entire fuel cycle, and all the acceptance criteria are met.

III.E. Accident Evaluation

III.E.1. Methodology and Results

In addition to the accidents evaluated in each of the other aspects, the accidental drop events were addressed.

1. Accidental Drop Events

The postulated drop events consider drop of one fuel assembly from a height of 6 feet above the top of the storage cells. Two worst case scenarios were evaluated.

The first scenario postulates that one fuel assembly drops vertically and impacts the top of a rack cell. The evaluation shows that the top of the impacted region undergoes localized plastic deformation. The maximum depth of plastic deformation is limited to 11 cm, which is less than the 34.5 cm from the top of the active fuel region to the top of the rack. Therefore, the damage does not extend into the active fuel region of the stored fuel.

The second scenario postulates that one fuel assembly falls through an empty storage cell impacting the rack baseplate. The evaluation shows that this scenario results in a maximum plastic strain of 26.2% in the baseplate, which is much smaller than the failure strain 40% of the baseplate material. Since the pool floor can maintain its overall integrity and the liner is not breached, there will be no abrupt or uncontrollable loss of water from the pool.

The evaluation shows that the drop events yield localized damage within the design limits.

2. Heavy Load Considerations for Rack Installation

NUREG-0612 provides guidelines to ensure safe handling of heavy loads by prohibiting load travel, to the extent practicable, over spent fuel assemblies, over the core, and over safety-related equipment.

There are two cranes in the fuel building: the cask crane and the fuel building crane. The cask crane is a single-failure-proof crane with sufficient capacity (150 tons) to place casks within the CLP. Because of physical travel limits of the cask crane (1,600 mm from the east end of the CLP), which prevent the main hook from reaching the center lines of two racks (B01 and B06 for Unit 1), the cask crane cannot be used to install all of the racks within the CLP. Consequently, the TPC has proposed to use the non-single-failure-proof fuel building crane (10 tons) for the rack installation, and to use the interlocks to prevent the fuel building crane from moving over the SFP to meet the "defense-in-depth" approach guidelines of NUREG-0612.

III.E.2. Key Safety Issues/Additional Analyses

TPC was requested by the AEC to address the effects of the rack drop. To lessen the impacts of the rack drop, the rack installation process regarding the usage of the two cranes has been revised as: (1) the cask crane is used to lower racks into the CLP to prevent the adverse effects of the rack drop from a high elevation, and (2) the fuel building crane is used to lift racks from the pool floor and move the racks horizontally with a limited lift height (15 cm) above the pool floor for rack positioning and fine-tuning.

IV. CONCLUSIONS

Expedited resolution of the key safety issues/additional analyses discussed above is extremely crucial to the timely approval of this project. The important experiences worth mentioning include:

1. Limit the stored fuels to old spent fuel assemblies is important for lessening the adverse impacts such as localized boiling and excessive exposure rate.
2. Exposure rate at the outer surface of the piping penetrating through the wall could exceeds the 0.05 mSv/hr limit for "Radiation Area", and might require administrative action.
3. Adjustment to the rack installation process regarding the usage of the two cranes in the fuel building is required due to their limitations: the single-failure-proof cask crane is used to lower racks into the CLP and the non-single-failure-proof fuel building crane is used for rack positioning and fine-tuning.

For Kuosheng Unit 1, the rack installation work was completed in May, and the unit was back online in June 2017. The successful implementation of this contingency project is extremely critical for achieving a stable power supply in Taiwan. The electricity operating reserve has fallen below the “red” alert (900 MWe) several times during last summer. Without this contingency project, Taiwan would have suffered the extremely adverse consequences of electricity rationing.

ACKNOWLEDGMENTS

The author would like to express his sincerest appreciation to Director Fu-Tien Chien and Deputy Directors (Alex Lin, Y. H. Hsu, and C. C. Chen) of the Department of Nuclear Generation, TPC, for the guidance and support. Appreciation is extended to C. H. Wu, H. K. Liu, and many other engineers at the Kuosheng site and the TPC headquarter, as well as T. Y. Liao and R. H. Tsai of PECL, for the extensive efforts on addressing the extremely complicated issues.

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附件二

ANS

PBNC 2018

PACIFIC NUCLEAR COUNCIL PNC

2018 Pacific Basin Nuclear Conference

Sustaining and Advancing Nuclear Energy

Safety Analyses for Installing Fuel Storage Racks in Cask Loading Pool at Kuosheng Nuclear Power Station

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PhD, PE

Taiwan Power Company

OVERVIEW

- Introduction
- General Description
- Safety Analysis and Key issues
- Conclusions

Introduction (1/2)

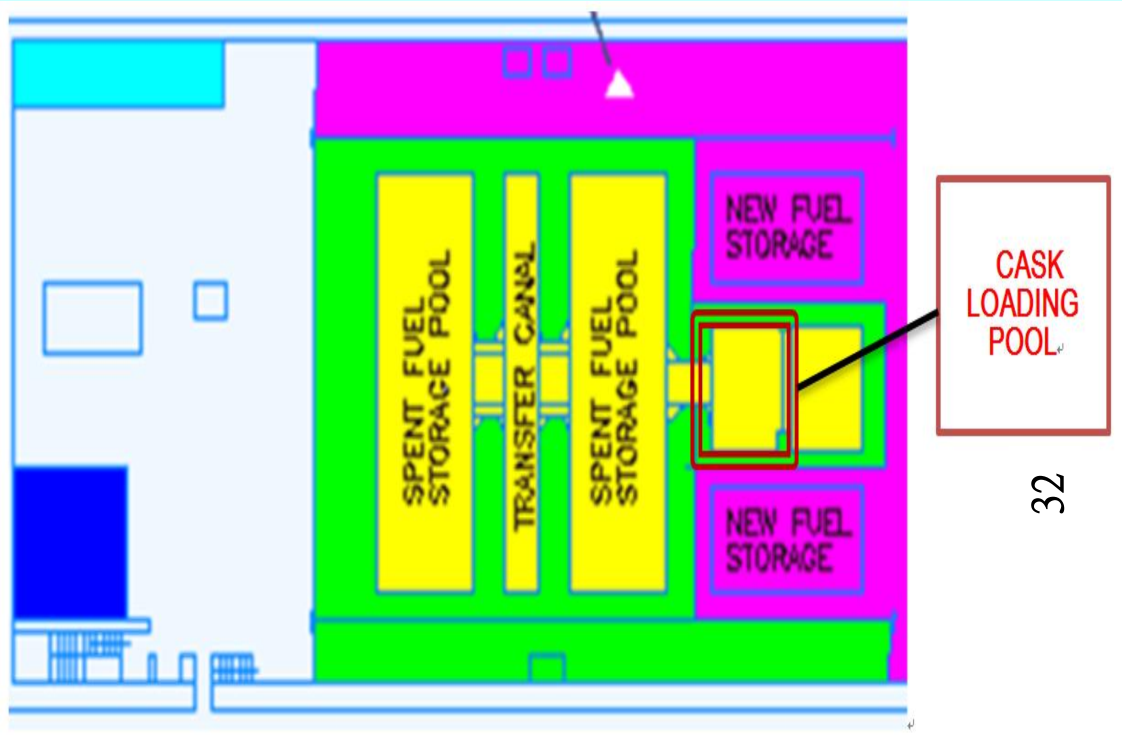
- This rack installation operation is a contingency project for Taiwan Power Company to maximize contribution of nuclear power in Taiwan.
- Kuosheng Nuclear Power Station consists of two BWR-6 units rated at 3001 MWt (~1020 MWe). Each unit has a Spent Fuel Pool (SFP) and a Upper Fuel Pool (UFP), however, UFP is not allowed for storage of spent fuel when the reactor is critical.
- The second SFP reracking project in 2004 adds six new racks to the SFP, and the total cell count is increased to 4398 cells.
- SFP for Kuosheng Unit 1 had practically no unfilled cells for discharged fuel such that further operation was prohibited from November 30, 2016 due to inability to refuel.
- TPC had planned to move some spent fuel to a dry storage facility. However, final approval for facility is blocked by local government.

Introduction (2/2)

- After a careful evaluation of the possible measures to restore refueling ability (mainly using the UFP or sending spent fuel abroad), TPC has elected to expand the storage capacity by installing four 11 by 10 storage racks from Lungmen in Cask Loading Pool (CLP). The additional 440 cells could allow operation for two more fuel cycles.
- Spent fuels allowed to be stored are limited to those offloaded from the first to the forth fuel cycles (decayed for more than 29 years).
- The effects on all aspects specified in AEC's review guidance are evaluated. In addition, relevant subjects are thoroughly addressed.
- Safety Analysis Report was approved on April 6, 2017. For Kuosheng Unit 1, the rack installation work was completed in May and the unit was back online in June 2017, just in time for peak electricity demand.

General Description - Pools and Racks (1/2)

- The SFP consists of two discrete pools, and the CLP is adjacent to the SFP.
- The second SFP reracking only adds six new rack modules supplied by ENSA. Except a specialty rack, all the existing rack modules supplied by Holtec from first reracking continue to be used.



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Pools and Racks (2/2)

- The Lungmen racks are also supplied by ENSA. The minimum Boral panel B-10 areal density (0.020 g/cm^2) is the same as the existing ENSA racks, and is much higher than that of the existing Holtec racks (0.012 g/cm^2). The cell pitch (167 mm) is substantially larger than that of existing racks (159.5 mm).
- Various pipings and other protrusions on the pool wall are removed. Existing stiffening frame around Lungmen racks is supplemented with a displacement restraint to provide horizontal support.

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Safety Analysis Report

- The scope of evaluations includes the five technical aspects specified in the AEC's review guidance:
  1. Criticality
  2. Thermal-Hydraulic
  3. Structure (Mechanical, Material and Structural)
  4. Radiological
  5. Accident
- Additionally, Rack transportation, installation, emergency plans, recovery plans, neutron absorber panel test plans, applicability of the post-Fukushima safety enhancement measures, and the integrity of the CLP are thoroughly addressed.

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Licensing Activities (1/2)

- The SAR was formally submitted on August 18, 2016. The AEC review process was separated into two stages: (1) acceptance review, and (2) detailed technical review.
- For the acceptance review, 21 questions were raised on the completeness of the report. After all the questions clarified, the report was accepted for the detailed review on September 20, 2016.
- For the detailed review, a review task force consists of scholars and experts of various aspects (including 11 external members and 13 AEC members) was organized to perform a very thorough review. The task force was separated into three functional groups:
  A. Accident/Criticality/Thermal-Hydraulic
  B. Structure/Material
  C. Radiological/Waste Treatment

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Licensing Activities (2/2)

- A joint review meeting for the three groups was conducted on September 21, 2016, and subsequently 10 review meetings for the individual group were conducted. Four rounds of RAI (with 62/85/40 questions for Groups A/B/C) were raised by the AEC.
- Some of the review questions required rather extensive efforts to address adequately, and follow-up questions to the initial questions were raised if the issues were not considered fully clarified. With the cooperative efforts by the TPC team, all the issues were fully clarified and SAR was approved on April 6, 2017.

Safety Analysis and Key issues - Criticality Analysis

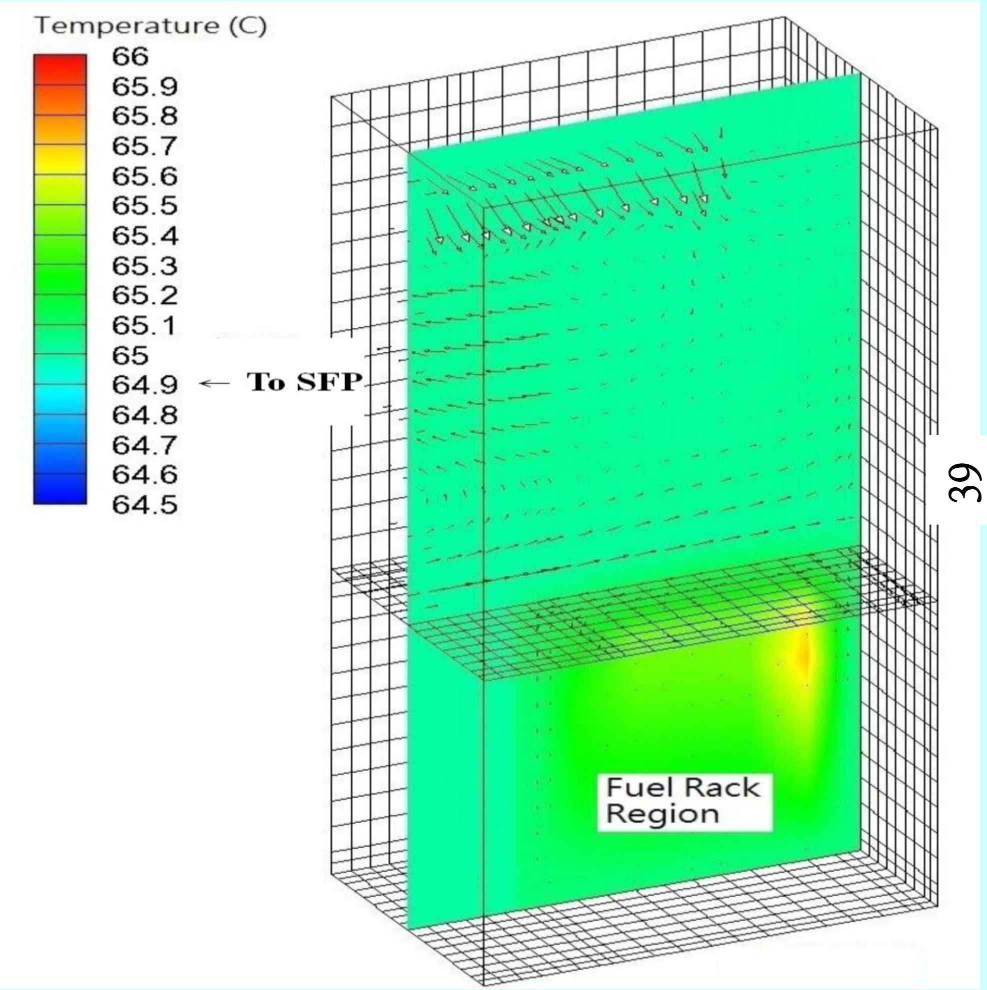
- Criticality analyses were performed with CASMO-4 and MCNP5 for all the lattices of two fuel types: GE8x8-2 and ANF8x8-2. The effects of abnormal and accident conditions were evaluated.
- The maximum neutron multiplication factor of the 9 lattices is 0.807, which is far less than the 0.95 limit. The maximum effect of abnormal and accident conditions is 0.0035.
- Since the Boral panel B-10 areal density and the cell pitch for Lungmen racks are more conservative with respect to criticality than those for existing racks, there is no real safety concern.
- TPC was requested to perform CASMO-4 calculations for the racks from Lungmen, the first reracking and second reracking. The results confirm that multiplication factor for Lungmen racks is indeed lower.

Thermal-Hydraulic Analysis - Methodology and Results

- The analysis evaluates: (1) maximum pool temperatures, (2) maximum local water temperatures, and (3) time available for corrective actions after loss of forced cooling. Since the increase in the heat load (~2%) is quite low, the effects are insignificant.
- The calculated maximum pool temperatures remain below the acceptance criteria, and the maximum local water temperature is lower than the saturation temperature. The time-to-boil is greater than 11 hours and time for water to drop to a level of 3 meters above top of fuel bundles is greater than 87 hours, thus, a sufficient time is available for corrective actions.

CLP Water Temperature Distribution

- There is no forced cooling flow injected into the CLP, and the decay heat load is removed by natural convection to the SFP through a gate (Gate 3) connecting the two pools. However, bottom of Gate 3 is approximately 17 feet above bottom of the CLP, and it was suspected that this might impede cooling of the CLP.
- GOTHIC was used to perform the analysis for storage of old spent fuel. The variation in temperature is only ~0.7°C, thus, Gate 3 could allow sufficient natural convection and localized boiling will not occur.



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Abnormal Refueling with Full Core Discharged to UFP

- FSAR requires that **elements that have decayed the longest in the spent fuel pools** may be moved into the upper pool storage racks during a full core outage, thus, “abnormal refueling case with full core discharged to UFP” is not considered an applicable case.
- Nonetheless, the maximum UFP water temperature for abnormal refueling case with the highest temperature was estimated using the relationship that the temperature rise is proportional to the heat load, and is inversely proportional to the coolant flow rate. With the increased heat load, maximum temperature estimated is 74.3 °C, still far below the limit value of 100 °C.

Structure Analysis - Methodology and Results (1/2)

- Structure analyses include two main parts: (1) rack dynamic analyses and structural evaluation, and (2) fuel building and CLP structural integrity evaluation.
- Structure analyses are complicated by post-Fukushima requirements and new finding for the nearby Shanchiao fault. The Safe Shutdown Earthquake in the FSAR is 0.4g, and the Operation Basis Earthquake is one half of the SSE. The effects of the new finding are considered:

- For newly added racks, a 0.67g Design Basis Earthquake is used

- For existing structure, 0.4g DBE and 0.2g OBE in the FSAR are used, and Seismic Margin Assessment methodology is used to address the Beyond Design Basis Earthquake of 0.67g.

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Structure Analysis - Methodology and Results (2/2)

- **Rack Dynamic Analyses and Structural Evaluation:** The analyses were performed in accordance with SRP 3.7.1 and 3.8.4. All the calculated stresses meet the ASME requirements.
- **Fuel Building and CLP Structural Integrity Evaluation:** The seismic impact of the additional fuel bundles and racks is very limited, all major structural components still meet the requirements. SMA High-Confidence of Low Probability of Failure analysis was also performed for the fuel building, and the criteria are met.
- **SFP/CLP Integrity Evaluation:** performed in accordance with EPRI-1025287, and results show that uncovering of the fuel bundles during normal operation and refueling outage will not occur within 72 hours.

Key Safety Issues/Additional Analyses

1. Total duration of time history : The value used is 24 second, which meets the SRP 3.7.1 requirement of at least 20 seconds.

- Since the 24 second duration is shorter than that experienced previously in Taiwan, a sensitivity analysis was requested.
- The calculated maximum stresses for a sensitivity case with 48 second duration are bounded by base case with 24 second duration.

2. Damping value: The 4%value used for SSE is consistent with SRP 3.7.1 and RG 1.61, but not consistent with 3% value specified in FSAR.

- The analyses meet AEC/NRC requirements and use of 4% is adequate.
- A sensitivity case was performed with a 3% damping, and the calculated maximum stresses meet the ASME requirements.

3. Safety margin with respect to fuel damage : Safety factor is ~7.7.

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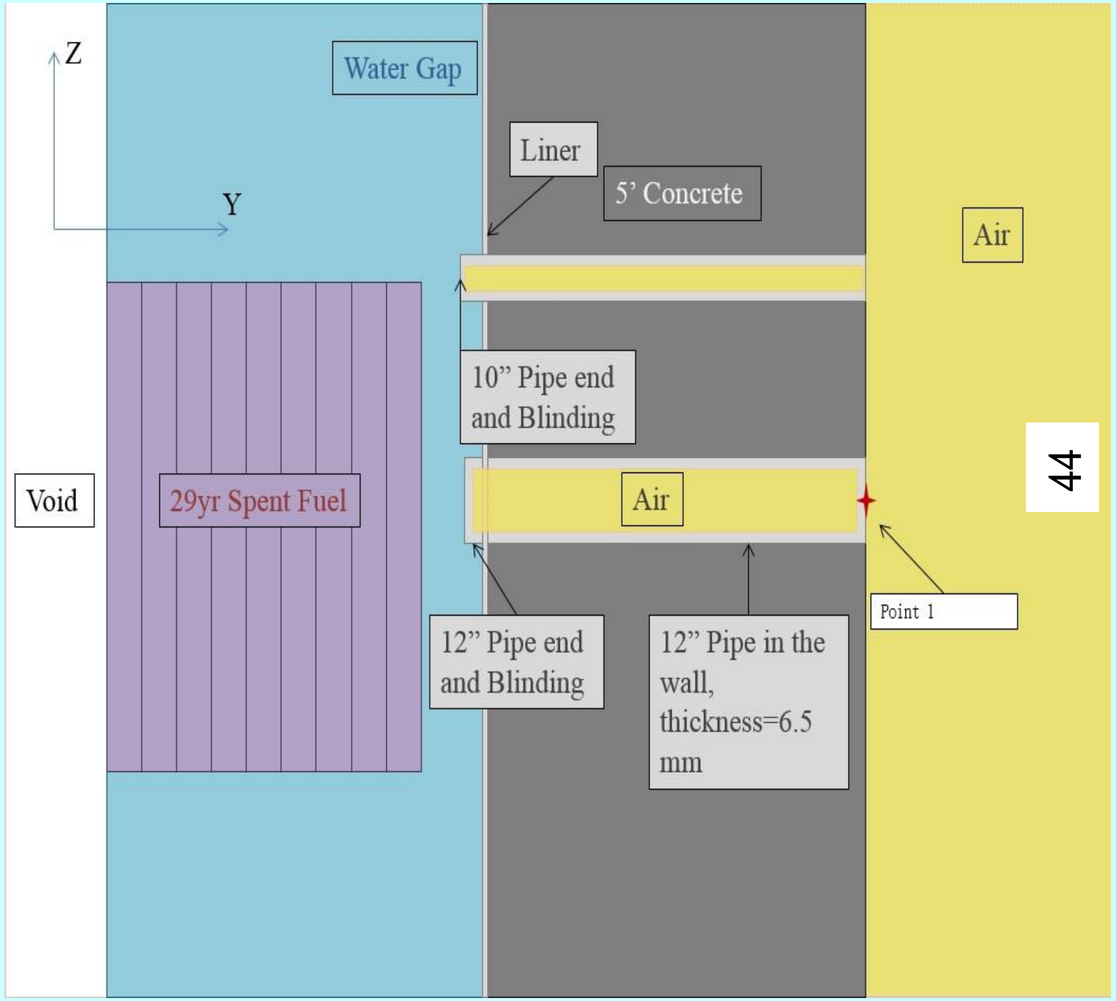
Radiological Evaluation - Methodology and Results

- The exposure rates from the SFP water, the occupational exposures, the generation of radioactive waste, and accident conditions were evaluated.
- The calculated exposure rate at the outer surface of the North wall, 0.0514 mSv/hr, exceeds the 0.05 mSv/hr limit for "Radiation Area" due to the 12-inch piping penetrating through the wall. Any area with a measured radiation level in excess of 0.05 mSv/hr will be classified and posted as "Radiation Area".



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Key Safety Issues/Additional Analyses

- The exposure rates from the SFP water, the occupational exposures, and offsite doses for loss of cooling to the SFP were originally evaluated with radionuclide concentrations from a single measurement. TPC was requested to address of the effect of concentrations variation during a fuel cycle.
- The relevant calculations were re-performed with maximum concentrations for the entire fuel cycle, and all the acceptance criteria are met.

Accident Evaluation - Methodology and Results

1. Accidental drop events: Two worst case scenarios were evaluated:

- The first scenario postulates that one fuel assembly drops vertically and impacts the top of a rack cell. The evaluation shows that the top of the impacted region undergoes localized plastic deformation. The maximum depth of deformation is limited to 11 cm, which is less than the 34.5 cm from top of active fuel region to top of the rack. Therefore, the damage does not extend into the active fuel region.
- The second scenario postulates that one fuel assembly falls through an empty cell impacting the rack baseplate. This scenario results in a maximum plastic strain of 26.2% in the baseplate, which is smaller than the failure strain 40%.

Accident Evaluation - Methodology and Results

2. Heavy Load Considerations for Rack Installation

- NUREG-0612 provides guidelines to ensure safe handling of heavy loads by prohibiting load travel over spent fuel assemblies.
- There are two cranes in the fuel building: the cask crane and the fuel building crane. The cask crane is a single-failure-proof crane with sufficient capacity (150 tons) to place casks within the CLP. Because of physical travel limits of the cask crane, which prevent the main hook from reaching the center lines of two racks, the cask crane cannot be used to install all the racks within the CLP.
- TPC proposed to use the non-single-failure-proof fuel building crane (10 tons) for the rack installation, and to use the interlocks to prevent the fuel building crane from moving over the SFP.

Key Safety Issues/Additional Analyses

- TPC was requested by the AEC to address the effects of the rack drop. To lessen the impacts of the rack drop, the rack installation process regarding the usage of the two cranes has been revised as:

  (1) the cask crane is used to lower racks into the CLP to prevent the adverse effects of the rack drop from a high elevation, and

  (2) the fuel building crane is used to lift racks from the pool floor and move the racks horizontally with a limited lift height (15 cm) above the pool floor for rack positioning and fine-tuning.

CONCLUSIONS

- Expedited resolution of the key safety issues/additional analyses discussed is extremely crucial for the timely approval of this project.
- Important experiences include:

- Limit the stored fuels to old spent fuel assemblies is important
- Exposure rate at outer surface (piping penetration) could exceed limit
- Optimized use of SF-proof cask crane & NSF-proof FB crane is desired

- The timely implementation of this contingency project is extremely critical for achieving a stable power supply in Taiwan. The electricity operating reserve has fallen below the "red" alert (900 MWe) several times during last summer. Without this project, Taiwan would have suffered the extremely adverse consequences of electricity rationing.

Thanks

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