INER-F0544 **INER-F0544**

出國報告（出國類別：實習）

赴美國史蒂文生公司研習核電廠結構地震動態分析與安全審查技術-國外實習報告

服務機關：核能研究所
姓名職稱：劉如峰　副工程師
派赴國家：美國
出國期間：100 年 10 月 15 日~100 年 12 月 10 日
報告日期：101 年 1 月 3 日

摘　要

此次實習主要配合「赴美國研習核電廠結構動態分析與安全審查技術」執行內容，前往美國史蒂文生公司(Stevenson & Associate, S&A)，研習核電廠結構地震動態分析與安全審查技術。此行獲得美國方面於核電廠耐震餘裕安全評估之考量條件與相關作法，以作為我國評估該設施安全性之重要參考，並透過研習核電廠耐震餘裕安全評估之分析架構與執行程序，提供適用於我國核能電廠之耐震餘裕安全評估，作為我國相關技術之重要基礎。本次實習期間，自 100 年 10 月 15 日起，至 100 年 12 月 10 日止，共為期 57 天。主要研習方向為核電廠結構地震動態分析與安全審查之理論與技術，應用相關分析方法評估核能電廠之安全性，並研討美國核電廠耐震餘裕安全評估技術之發展與應用。透過與該公司專家學者之指導與交互討論，並蒐集有關的技術資料，以了解及增進核電廠耐震餘裕安全評估之技術及經驗。為確保在評估基準地震下安全停機有關之結構、設備及組件，應仍保有高度的可信度可持續安全運轉，並應用於我國核電廠組件完整性分析，以客觀之耐震餘裕分析技術，評估核電廠安全餘裕，以作為電廠遭遇各種假相情況之安全評估，並增進該設施的壽命與安全性。

關鍵字：核能結構完整性、地震動態分析、耐震餘裕分析

目　　次

摘　要

一、 目的

此次實習主要配合「赴美國研習核電廠結構動態分析與安全審查技術」執行內容，前往美國史蒂文生公司(Stevenson & Associate, S&A)，研習核電廠結構地震動態分析與安全審查技術。

史蒂文生公司(Stevenson & Associate, S&A) 成立於 1981 年，總公司位於美國波士頓近郊的屋本市(Woburn City)，是美國主要的核能工程服務公司之一，迄今已經三十餘年，另外於克里夫蘭、芝加哥及明尼蘇達等三處亦有其辦公室。S&A 公司服務對象由初期的 Power (nuclear and fossil) Structural Engineering 及 Seismic 分析，擴展至 Process 及 Manufacturing Industries 的領域，服務項目包括 Mechanical、Piping 及 System Engineering 和 Software Development，亦提供有 Valves、In-service Inspection 與 Testing 方面的服務。S&A 公司在核能電廠主要專業工作，為核能電廠組件設計與分析、ASME 鍋爐與壓力容器法規(Boiler & Pressure Vessel code)顧問、核能管路設計分析及結構耐震評估等。在核能管路設計及分析方面，S&A 公司的主要經驗包括依據 ASME Sec. III 執行核能一、二及三級管路的設計分析與疲勞分析，管路支撐設計，管路耐震分析及反應頻譜，蒸汽管路與水鍾損壞肇因分析與高溫管路系統分析等。因應美國 911 事件之後核能電廠及其他公共建設結構防爆安全防護要求的逐漸提升，S&A 公司近年來也協助有關單位執行結構物防破壞及防恐相關的分析與評估工作，包括結構爆破強度評估、改善補強建議與現場視察(Site Walkdown)等。該公司副總裁 Dr. T. M. Tseng，近年曾執行本所執行「核二廠時限整體安全評估技術服務」案「核一廠核能一級管路應力分析與技術諮詢」，具有相當豐富的核能結構、管路及組件的應力分析、電廠耐震與振動分析經驗，尤其針對核能一級組件設計與分析，擁有多年實務經驗與研究成果發表，可為本所學習典範。S&A 公司另於 1984 年成立一獨立之子公司：Vibration Engineering Consultants, VEC，專門為高科技製程(manufacturing)，如半導體、奈米科技、電子等和其他 Vibration-Sensitive Facilities 如醫院、研究實驗室等，提供 Vibration Analysis 及 Design Services。

因應 100 年 3 月 11 日日本東北發生芮氏規模 9.0 強震（日本官方名稱訂為「平成 23 年東北地方太平洋沖地震」)，引發海嘯造成重大核災事故，S&A 公司刻正執行台灣電力公司「營運中核能一廠耐震安全餘裕評估計畫」。美國核能電廠在營運後才發現，新事證的地震危害度大於建廠時的考量，例如美國東北部 Maine Yankee 核能電廠 SSE 為 0.1g，但於 1982 年 1 月 9 日因加拿大規模 5.75 的地震，超過電廠設計基準，電廠即進行地震分析計畫（Seismic Analysis Program)。另外，美國地質調查所於 2004 年完成美國中部及東部地震危害度評估初步報告，在該二地區的部分核能電廠其受地震危害的程度，較原先設計階段所預估的要高，因此必須辦理耐震餘裕評估（Seismic Margin Assessment, SMA）。此行透過與該公司專家學者之指導與交互討論，並蒐集有關的技術資料，以了解及增進核電廠耐震餘裕安全評估之技術及經驗。為確保在評估基準地震下安全停機有關之結構、設備及組件，應仍保有高度的可信度可持續安全運轉，並應用於我國核電廠組件完整性分析，以客觀之耐震餘裕分析技術，評估核電廠安全餘裕，以作為電廠遭遇各種假相情況之安全評估，並增進該設施的壽命與安全性。已穫如下成效：

(一) 獲得美國方面於核電廠耐震餘裕安全評估之考量條件與相關作法，以作為我國評估該設施安全性之重要參考。

(二) 研習核電廠耐震餘裕安全評估之分析架構與執行程序，提供適用於我國核能電廠之耐震餘裕安全評估，作為我國相關技術之重要基礎。

本次實習期間，自 100 年 10 月 15 日起，至 100 年 12 月 10 日止，共為期 57 天。主要研習方向為核電廠結構地震動態分析與安全審查之理論與技術，應用相關分析方法評估核能電廠之安全性，並研討美國核電廠耐震餘裕安全評估技術之發展與應用。

二、 過程

此次實習共計 57 天，由 100 年 10 月 15 日至 12 月 10 日。實習地點為美國史蒂文生公司，研習核電廠結構地震動態分析與安全審查技術，透過其研究工作之參與、文獻之閱讀、分析程式之執行，與數據資料之整理，瞭解核電廠耐震餘裕安全評估之基本理論、背景與應用方法，以增進我國核電廠耐震餘裕安全評估等相關技術之發展。

行程及工作日誌大要如下：

日期	行程	公差地點	工作內容
100.10.15(六)	台北→屋本市	屋本市	去程
100.10.16(日)		屋本市	休息及準備、整理資料
100.10.17(一) | 100.12.08(四)		屋本市	研習核電廠結構地震動態分析與安全審查技術
100.12.09(五) | 100.12.10(六)	屋本市→台北		返程

<u>10 月 15 日，星期六</u>

早晨自桃園機場搭乘中華航空 CI 20 班機出發，先到日本關西國際機場加油，於美東當地時間中午時分到達紐約市約翰甘迺迪國際機場。順利完成入境手續之後，緊接著轉搭乘達美航空 DL 2417 班機飛往波士頓羅根國際機場。由於抵達當地時間已是週末晚上，便自行搭計程車前往下榻旅館投宿。

<u>10 月 16 日，星期日</u>

因超過 24 小時之長途飛行與轉機等待，加上尚須調適時差，本日外出購買生活所需物品後，便回旅館休息以儲備體力。

<u>10 月 17 日，星期一</u>

S&A 公司 Mr. Mo-Hwa Wang 至旅館接送前往公司。由於 S&A 公司屬於私人企業，並不需要瑣碎而煩雜的報到手續，並且直接發給門禁卡以便爾後自行出入。當日早晨與 S&A 公司總裁 Mr. Walter Djordjevic 短暫面談之後，Mr. Mo-Hwa Wang 帶領至辦公室，介紹公司相關同事彼此認識，包括資深工程師 Mr. John Holland、Mr. Joshua Hart 及 Todd Radford 等。當日部分員工前往外地出差，日後則陸續認識了副總裁 Dr. T. M. Tseng、專案經理 Dr. Keith Xu、資深工程師 Dr. Yi-Lun Chu 與 Mr. Sung-June Kim 等。稍作寒喧後，繼續由 Mr. Mo-Hwa Wang 帶領大概介紹工作環境，並且參觀 S&A 公司設備後，隨即開始安排工作位置與相關事宜。

<u>10 月 18 日，星期二 至 12 月 8 日，星期四</u>

展開為期八週之各項研究工作，主要針對地震機率式風險評估（Seismic Probabilistic Risk Assessment, SPRA）和地震易損度評估（Seismic Fragility Analysis）等工作上，引入耐震餘裕（Seismic Margin Assessment, SMA）概念並將其與地震易損度之關係予以結合，此一關係為本報告進行耐震餘裕評估之基礎。同時以台電公司金山核能一廠為例，參考其 1990 年之機率式風險評估報告中有關結構地震易損度評估部分進行整理，並摘錄原地震易損度評估所得之耐震餘裕結果，最後依據該報告之結果，利用美國最新耐震餘裕評估作法，評估修正核一廠結構之耐震餘裕度。

目前推行性能基準是著眼於風險控制，這與美國核能體系發展風險告知有關，藉由爐心熔毀頻率的機率風險評估機率模式計算，了解各核能電廠的風險，以及內部結構系統組件的相對重要性，使得管制資源能較為合理分配到對安全較為重要的事務。性能基準的法規著重在其結果是所管制單位所期望的可量化結果，而非僅是制定過程或是要求技術，由結果面進行管制。地震機率式風險評估已成功應用於全球超過 50 座核能發電廠，其流程如圖一所示。其主要目的為瞭解地震事件所可能引致之爐心損壞可能性，並找出造成地震風險的主要因素。地震機率式風險評估中所需的必要分析包含：

(一) 地震危害度分析（Seismic Hazard Analysis）

(二) 地震易損性分析（Seismic Fragility Analysis）

(三) 意外序列分析（Accident Sequence Analysis）

(四) 風險量化（Risk Quantification）

其中，地震危害度分析(Seismic Hazard Analysis)本質上在進行地震災害影響的預測工作，期望建立震源-路徑-場址之波傳行為，以獲得工程設計可使用的廠址地震波或反應譜。由於地震發生地點難以預期，僅能由地震紀錄預測未來可能發生位置，作為危害度分析之震源，方能進行後續計算。為了瞭解地震的回歸期，須研究斷層或是特定地質構造的地震成因與趨勢，或採地震發生率的統計方法解決，由機率式地震危害度分析產生危害度曲線，即是目前建立設計地震回歸期的方法。另外，過去在決定設計地震時，僅考慮地震在震源-路徑-廠址的效應，係屬於地震科學的評估結果，而非工程防災的結果，且沒有整體設施風險的觀念在裡面。工程防災設計的結果可用損壞曲線為代表，透過與危害度曲線之間的關係，則可將設施的性能表現併入設計地震。地震易損性分析（Seismic Fragility Analysis）主要是在定義單一結構或元件在指定之地震參數下的條件破壞機率。目的為找出單一結構或元件之地震強度（Seismic Capacity）及其變異性。變異性主要來自預測地震輸入、材料強度、分析方式，分析採用參數等所隱含的隨機性（randomness）及不確定性（uncertainty）。不論何種結構或破壞模式的地震易損性曲線，皆利用以下參數決定

- Median Ground Motion Capacity, A_m
- Logarithmic Standard Deviation for Randomness, β_R
- Logarithmic Standard Deviation for Uncertainty, β_U

決定上述地震易損性曲線參數時，針對的不同結構必須分別考慮以下條件：

<u>Structures</u>

<u>Capacity</u>

Strength（yield or ultimate）

Inelastic energy absorption

<u>Response</u>

Ground response spectra

Foundation - structure interaction（including soil structure interaction, deconvolution & incoherence）

Damping

Frequency

Mode shape

Torsional coupling

Mode combination

Time history simulation

Earthquake component combination

<u>Equipment</u>（qualified by analysis）

Equipment capacity

Strength（yield or ultimate）or test capacity

Inelastic energy absorption

Building structure response

Equipment response

Qualification method

Damping

Frequency

Mode shape

Mode combination

Earthquake component combination

<u>Equipment</u>（qualified by testing）

Test capacity

Building structure response

Equipment factors

Response clipping

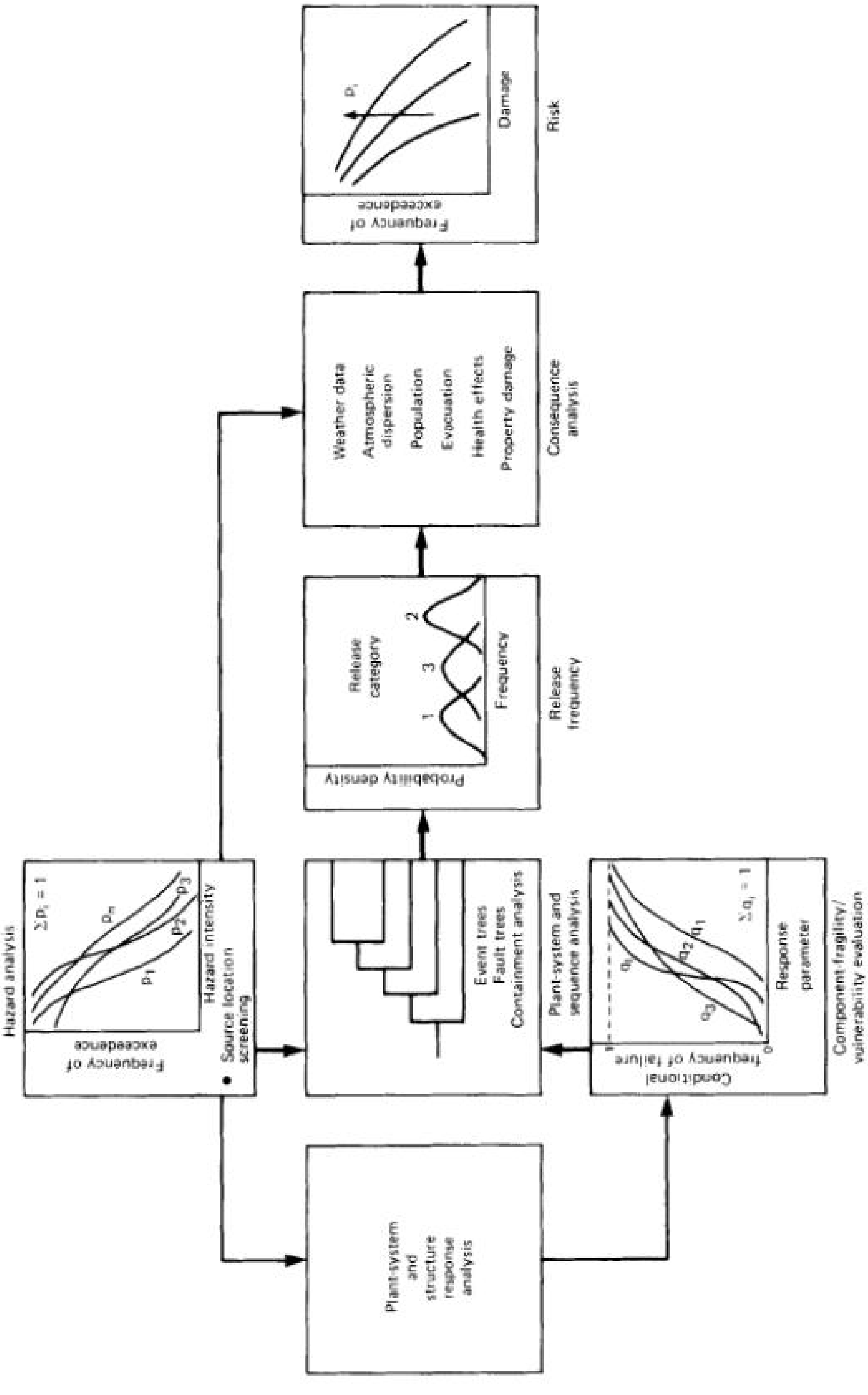
Capacity increase and demand reduction

Cabinet amplification

Multi-axis to single-axis conservatism

Broad frequency input spectrum device capacity

地震易損性可配合電廠邏輯模型（Plant Logic Model）或故障樹（Fault Tree），日後用於地震機率式風險（Seismic Probabilistic Risk Assessment）評估，可以估算整廠破壞機率。



圖一、地震機率式風險評估

耐震餘裕評估（Seismic Margin Assessment, SMA）主要是在定義電廠在超過某一地震強度下可能會失去安全停機能力，特別是導致爐心熔毀。目前較為通用的耐震餘裕度量參數為高信度低損害率（High-Confidence-Low-Probability-of-Failure, HCLPF）之最大地表加速度 PGA。HCLPF 是由相對保守的定量方法所得出，而地震易損性分析（Seismic Fragility Analysis）與耐震餘裕（Seismic Margin or HCLPF）評估之結果其實完全相容，兩者之差異性僅在於地震易損性分析計算出結構抗震度之完整概率分布曲線，包含中位數（Median）及對數標準差（Log Standard Deviation），而耐震餘裕僅為一點估計。在機率觀點上，耐震餘裕代表的是在95%的信心水準下，若地表強度超過所求得知 HCLPF，其破壞機率小於 5%，兩者合併破壞機率約為 1%，保守程度相當於一般結構設計規範。地震機率式風險評估與耐震餘裕評估主要不同在於後者僅估算電廠保守耐震度，並未卷積（Convolute）地震易損性與地震危害度（Seismic Hazard），因此無法計算全廠破壞機率。耐震餘裕評估的主要目的為確定電廠擁有足夠之餘裕以承受大於原設計地震(Safe Shutdown Earthquake, SSE)，並辨識出電廠耐震設計最弱之環節。根據 U.S. NRC 文件 NUREG-1407 [1] 評估基準地震（Review Level Earthquake, RLE）的選定建議，RLE 的地表反應譜可採用 NUREG/CR-0098 [2]中值岩石反應譜（Median Rock Spectrum），核一廠聯合結構廠房地基直接落於砂岩盤上。最大地表加速度（Peak Ground Acceleration, PGA）則依據 U.S. NRC SRM to SECY-93-087, Page 7, Item 17 [3]：

"The Commission approves the use of 1.67 times the Design Basis SSE for margin-type assessment of seismic events."

訂為 1.67 倍之設計地震地表加速度，$1.67 \times 0.30\ g = 0.50\ g$，未來當地震危害度評估完成時，配合 U.S. NRC, Generic Letter 2011-XX [4]，將使用 Regulatory Guide 1.208 [5] GMRS。

美國核能電廠亦有營運後才發現新事證的地震危害度大於建廠時的考量，例如美國東北部 Maine Yankee 核能電廠 SSE 為 0.1g，但於 1982 年 1 月 9 日因加拿大規模 5.75 的地震，超過電廠設計基準，電廠即進行 Seismic Analysis Program。另外，美國地質調查所於 2004 年完成美國中部及東部地震危害度評估初步報告，在該二地區的部分核能電廠其受地震危害的程度較設計階段所預估的要高，因此須辦理耐震餘裕評估，其理論基礎係採用 EPRI 報告 NP-6041-SL revision [6]作為耐震餘裕評估。我國核能電廠情況類似，應參照美國做法辦理，在考慮地質新事證的情況下，以 SMA 或 SPRA 的結果證明耐震安全性。因應近來公布山腳斷層及恆春斷層為第二類活動斷層，及民國 95 年 12 月 26 日恆春西南外海發生 2 起規模 7.0 之地震，核三廠測量到 0.165g 之地表加速度，為核能電廠運轉 30 年來最大值，這些新事證應經詳細調查後，將結果納入核能電廠的地震危害度的評估內，地震危害度曲線亦是隨後評估基準地震訂定的參考，直接影響耐震評估的結果。為求安全停機，相關之設備、組件及系統均需檢查評估以確保強震下仍能維持應有之功能，且依法規要求能維持 72 小時。所有需要進行評估的項目將由承包廠商專業人員與電廠現場運轉、維護人員討論後提出。評估挑選出之設備、組件及系統是否有足夠之 HCLPF，評估時接受準則應遵照 EPRI NP-6041 規定。由專業人員與電廠現場運轉、維護人員進行現場巡查，以確認前所條列出的設備、組件及系統是否為安全停機相關之設備、組件及系統，並確認所選定項目可以進行安全停機之計算評估。原設計之樓層反應譜未考量新地質事證，而樓層反應譜是為供設備耐震評估之用，因此，除考量新地質事證之影響外，另依現今對土壤-結構互制的概念成熟度，可據以製作樓層反應譜。最後再由耐震評估小組（Seismic Review Team, SRT）評估挑選出之設備、組件及系統是否有足夠之 HCLPF，評估時接受準則應遵照 EPRI NP-6041 規定。

為進行地震機率式風險評估，需進一步了解機率函數所採用的機率分佈函數，包含有：均勻分佈（Uniform Distribution）、常態分佈（Normal Distribution）、對數常態分佈（Lognormal Distribution）、偉伯分佈（Weibull Distribution）、邏輯分佈（Logistic Distribution）與強森 S_B 分佈（Johnson S_B Distribution）。針對以上分佈函數之機率密度函數（Probability Density Function, PDF）以及其累積分佈函數（Cumulative Distribution Function, CDF）作概略說明。

(一) 均勻分佈：係為其他連續分佈函數轉換之依據。均勻分佈之機率密度函數與累積分佈函數定義如下：

$$\text{PDF}:\ f_u(x\,|\,a,b)=\begin{cases}0 & ;x<a\\ \dfrac{1}{(b-a)} & ;a\le x\le b\\ 0 & ;x>b\end{cases}$$

$$\text{CDF}:\Pr(X\le x)=F_u(x\,|\,a,b)=\begin{cases}0 & ;x<a\\ \dfrac{x-a}{(b-a)} & ;a\le x\le b\\ 0 & ;x>b\end{cases}$$

其中，a 與 b 分別為函數之下界與上界。均勻分佈函數之平均值 $\mu=(a+b)/2$，變異數 $\sigma^2=(b-a)^2/12$。利用雙函數均勻分佈函數取樣 $U_i\leftarrow U(0,1)$ 時，可採用 Fortran 程式的亂數產生器（Random Number Generator, RNG）於區間內隨機取樣，該亂數於 32 位元電腦執行之最短週期為 2.3×10^{18}，該取樣法由 Rukhin 等學者於 2001 年測試並驗證其適用性。

(二) 常態分佈：機率密度函數與累積分佈函數定義如下：

$$\text{PDF}:\ f_N(x\,|\,\mu,\sigma)=\frac{1}{\sigma\sqrt{2\pi}}\exp\left[-\frac{(x-\mu)^2}{2\sigma^2}\right];\quad -\infty<x<+\infty$$

$$\text{CDF}:\Pr(X\le x)=\frac{1}{\sigma\sqrt{2\pi}}\int_{-\infty}^{x}\exp\left[-\frac{(x-\mu)^2}{2\sigma^2}\right]dx$$

$$=\frac{1}{\sigma\sqrt{2\pi}}\int_{-\infty}^{x}\exp\left[-\frac{(u-\mu)^2}{2\sigma^2}\right]du=\frac{1}{\sqrt{2\pi}}\int_{-\infty}^{z}\exp\left[-\frac{\xi^2}{2}\right]d\xi=\Phi(z)$$

常態分佈函數之平均值為 μ，變異數為 σ^2。

(三) 對數常態分佈：機率密度函數與累積分佈函數定義如下：

$$\text{PDF}:\ f_\Lambda(x\,|\,\mu_{\log},\sigma_{\log})=\begin{cases}0 & ;\quad x\le 0\\ \dfrac{1}{\sigma_{\log x}\sqrt{2\pi}}\exp\left[-\dfrac{(\ln x-\mu_{\log})^2}{2\sigma_{\log}{}^2}\right] & ;\quad 0<x<\infty\end{cases}$$

$$\text{CDF}:\Pr(X\le x)=\begin{cases}0 & ;\quad x\le 0\\ \dfrac{1}{\sqrt{2\pi}}\displaystyle\int_{-\infty}^{z}\exp\left[-\dfrac{\xi^2}{2}\right]d\xi=\Phi(z) & ;\quad 0<x<\infty\end{cases}$$

$$z=\frac{\ln x-\mu_{\log}}{\sigma_{\log}}$$

對數常態分佈的平均值 $\mu=\exp(\mu_{\log}+\sigma_{\log}{}^2/2)$，而其變異數可表示為 $\sigma^2=\exp(\sigma_{\log}{}^2)\times[\exp(\sigma_{\log}{}^2)-1]\times\exp(2\mu_{\log})$。

(四) 偉伯分佈：分為 a、b 與 c 三個參數，分別為位置參數（Location Parameter）、比例

參數（Scale Parameter）與形狀參數（Shape Parameter）。偉伯分佈之機率密度函數與累積分佈函數定義如下：

$$\mathrm{PDF:}\ f_W(x\mid a,b,c)=\begin{cases}0 & ;\ x\le a\\ \dfrac{c}{b}\left(\dfrac{x-a}{b}\right)^{c-1}\exp\left(-\left(\dfrac{x-a}{b}\right)^{c}\right) & ;\ 0<a,b,c<x\end{cases}$$

$$\mathrm{CDF:}\Pr(X\le x)=F_W(x\mid a,b,c)=\begin{cases}0 & ;x\le a\\ 1-\exp\left(-\left(\dfrac{x-a}{b}\right)^{c}\right) & ;0<a,b,c<x\end{cases}$$

該函數的平均值 $\mu=a+b\times\Gamma(1+\frac{1}{c})$，其中 $\Gamma(x)$ 為 Euler's Gamma Function。變異數 $\sigma^2=b^2\left[\Gamma\left(1+\frac{2}{c}\right)-\Gamma^2\left(1+\frac{1}{c}\right)\right]$。

（五） 邏輯分佈：機率密度函數與累積分佈函數定義如下：

$$\mathrm{PDF:}\ f_L(x\mid\alpha,\beta)=\frac{\exp\left[-\left(\dfrac{x-\alpha}{\beta}\right)\right]}{\beta\left\{1+\exp\left[-\left(\dfrac{x-\alpha}{\beta}\right)\right]\right\}}\ ;-\infty<x<+\infty$$

$$\mathrm{CDF:}\Pr(X\le x)=F_W(x\mid\alpha,\beta)=\frac{1}{1+\exp\left[-\left(\dfrac{x-\alpha}{\beta}\right)\right]}\ ;-\infty<x<+\infty$$

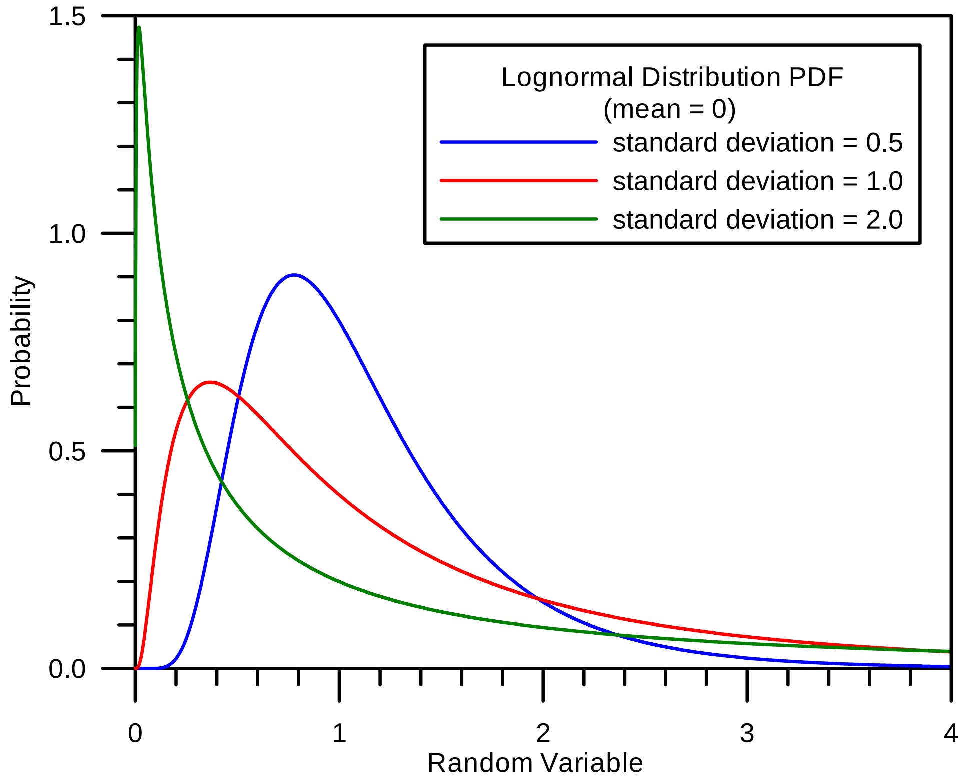
該分佈之平均值 $\mu=\alpha$，變異數 $\sigma^2=\dfrac{(\pi\beta)^2}{3}$。

（六） 強森 S_B 分佈：機率密度函數與累積分佈函數定義如下：

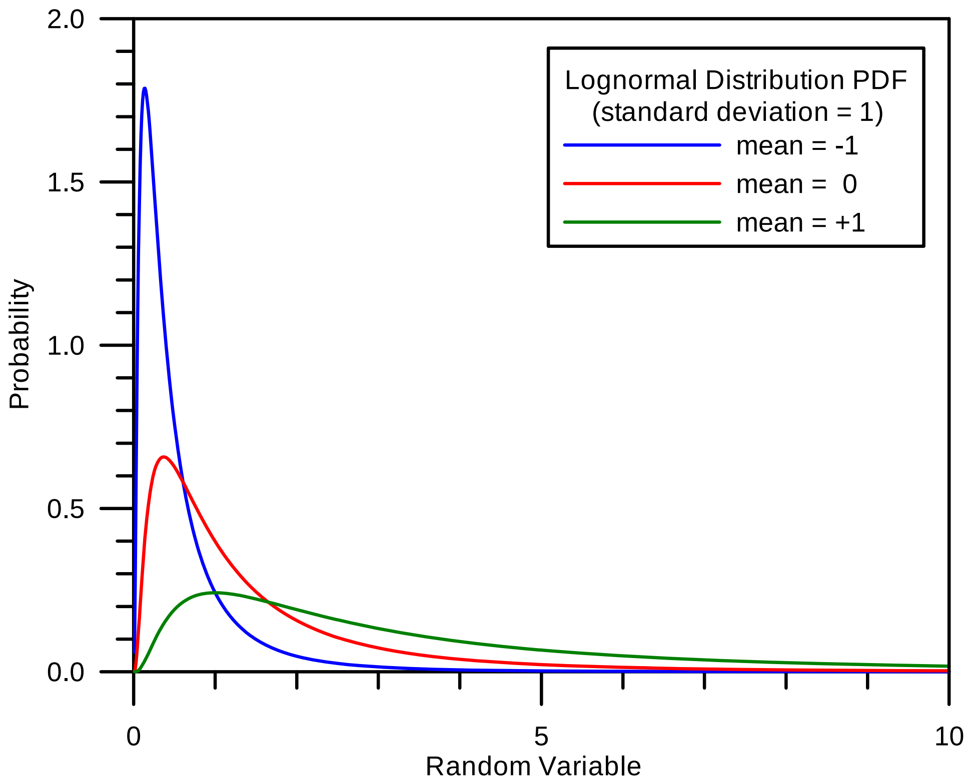
$$\mathrm{PDF:}\ f_{JS_B}(x\mid a,b,\alpha_1,\alpha_2)=\begin{cases}0 & ;\ \text{for } x\le a\ \text{ or } x\ge b\\ \dfrac{\alpha_2(b-a)}{(x-a)(b-x)\sqrt{2\pi}}\exp\left\{-\dfrac{1}{2}\left[\alpha_1+\alpha_2\ln(\dfrac{x-a}{b-x})\right]^2\right\} & ;\ \text{for } a<x<b\end{cases}$$

$$\mathrm{CDF:}\Pr(X\le x)=F_{JS_B}(x\mid a,b,\alpha_1,\alpha_2)=\begin{cases}0 & ;\ \text{for } x\le a\\ \Phi\left[\alpha_1+\alpha_2\ln(\dfrac{x-a}{b-x})\right] & ;\ \text{for } a<x<b\\ 1 & ;\ \text{for } x\ge b\end{cases}$$

針對以上機率密度函數的數值表示，除了一般統計分析軟體之外，亦可利用 Fortran 程式直接撰寫。如圖二所示，係相同的平均值針對不同的標準差，在 0 到 4 的區間製的對數常態分佈。圖三則是相同的標準差針對不同的平均值，在 0 到 10 的區間所繪製的對數常態分佈。再利用數值積分方法，可將圖二或圖三的機率密度函數分佈轉換為累積分佈函數。由於一般的機率密度函數都是屬於二次曲線，沒有複雜的奇異值可能發生不可積分的問題需要處理，針對二次曲線的分佈型式，可利用辛普森第二法則（Simpson's 2nd Rule）進行數值積分計算，係簡單又便捷的數值積分方法。當某一方程式需要積分時



圖二、對數常態分佈



圖三、對數常態分佈

$$I = \int_a^b f(x)dx$$

若利用梯形積分法（亦稱為辛普森第一法則），則其積分值 T_n 為

$$T_n = \frac{h}{2}[f(a) + 2f(x_1) + 2f(x_2) + \cdots + 2f(x_{n-1}) + f(b)]$$

其中，x_i=a+ih，i=0,1,…,n，h=(b-a)/n。一般而言，梯形積分法比較適合常數或一次函數，二次函數的積分值 S_n 則為

$$S_n = \frac{h}{6}[f(a) + 4f(x_{1/2}) + 2f(x_{1/2}) + 4f(x_{3/2}) + \cdots + 2f(x_{n-1}) + 4f(x_{n-1/2}) + f(x_n)]$$

其中，x_i=a+ih，i=0,1/2,1,3/2,…,n-1,n-1/2,n，h=(b-a)/n。如圖四所示，為平均值 0、標準差 0.5 之累積分佈函數，亦可解釋為對數常態分佈在 0 到 5 的區間所積分的結果。累積分佈函數在數學上的定義皆是在±∞的區間內，但是執行上只要依據函數特性取其夠大的積分空間，其累積分佈積分值即可趨近於 1，而不需要進行過多的數值計算。地震易損性評估一般皆是採用對數常態分佈，配合地震輸入、材料強度、分析方式，以及分析採用參數等所隱含的隨機性及不確定性，即可決定出地震易損性曲線。

利用數常態分佈之概念，可建立結構或設備的地震易損性曲線。依據 EPRI 報告 TR-103959[7]，亦可使用正規化方法求得地震易損性曲線，以有效降低數值積分的電腦計算時間。其方法係先尋求對數常態之累積分佈函數之正規化數據，如表一所示，其圖形分佈如圖五所示，再利用必需的中位數和平均值，即可內插求得累積分佈機率。

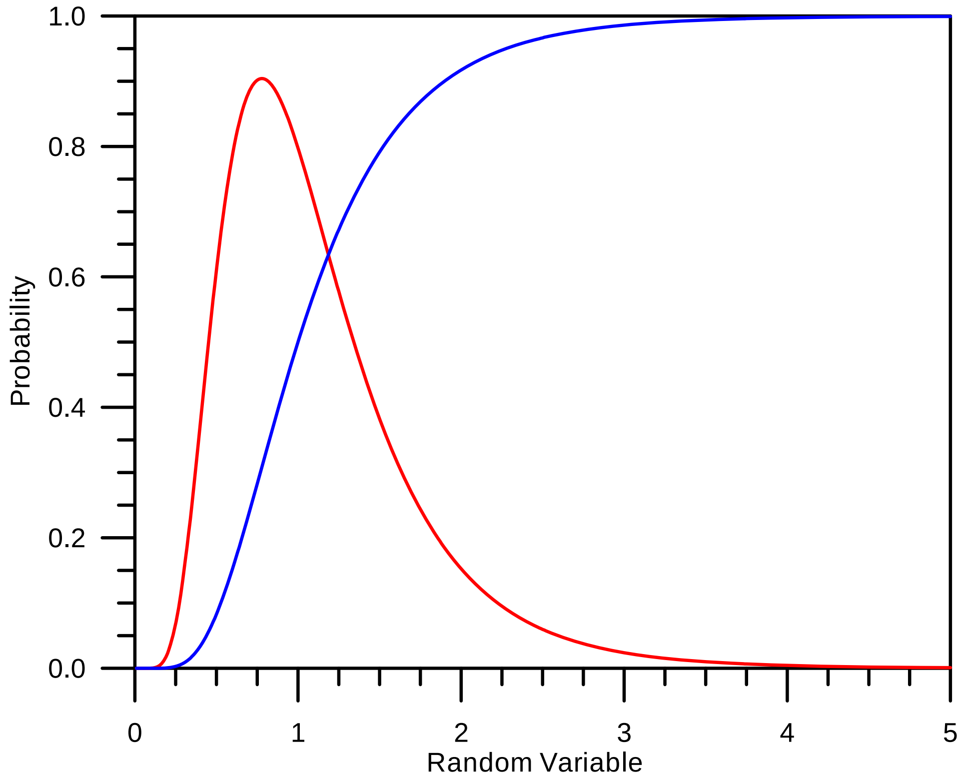
表一、正規化對數常態累積分佈函數

				σ_y		
z	F(z)	0.5	1.0	1.5	2.0	2.5
-3.0	0.001	0.22	0.05	0.01	0.00	0.00
-2.5	0.006	0.29	0.08	0.02	0.01	0.00
-2.0	0.023	0.37	0.14	0.05	0.02	0.01
-1.5	0.067	0.47	0.22	0.11	0.05	0.02
-1.0	0.159	0.61	0.37	0.22	0.14	0.08
-0.5	0.309	0.78	0.61	0.47	0.37	1.29
0.0	0.500	1.00	1.00	1.00	1.00	1.00
0.5	0.691	1.28	1.65	2.12	2.72	3.49
1.0	0.841	1.65	2.72	4.48	7.39	12.18
1.5	0.933	2.12	4.48	9.49	20.09	42.52
2.0	0.977	2.72	7.39	20.09	54.60	148.41
2.5	0.994	3.49	12.18	42.52	148.41	518.01
3.0	0.999	4.48	20.09	90.02	403.43	1808.04

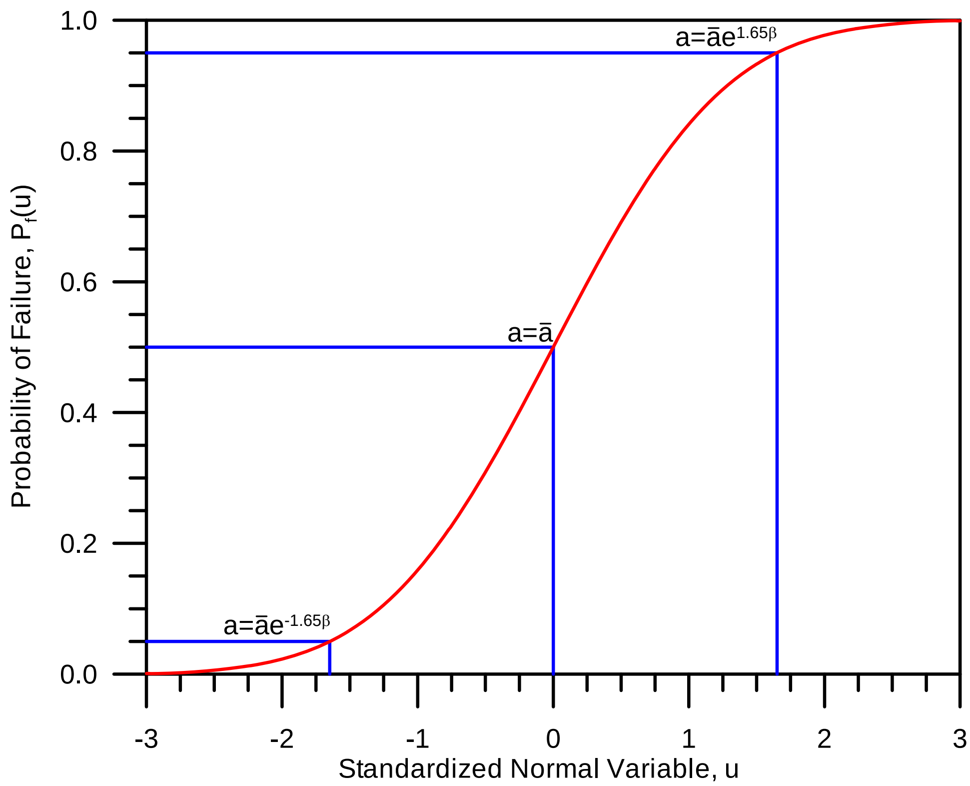
令

$$P_f(a) = \Phi(u)$$

其中，$P_f(a)$為任意常數 a 的失效機率，$\Phi(u)$為正規化常數 u 的失效機率。由於地震易損性評估係採用對數常態分佈所建立，故正規化常數 u 與常數 a、中位數 $\tilde{a}$ 、標準差 β 的關係，亦是對數型態分佈，可表示為



圖四、對數常態累積分佈函數



圖五、正規化對數常態累積分佈函數

$$u = \ln(a/\tilde{a})/\beta$$

因此，若能掌握中位數$\tilde{a}$、標準差β，即可透過常數 a 轉換為正規化常數 u，再內插求得正規化常數 u 的失效機率$\Phi(u)$。另外，亦可將上式轉換為

$$a = \tilde{a} \cdot e^{u\beta}$$

此式代表在已知某失效機率 P_f 下，經過逆轉換求得正規化常數 $u=\Phi^{-1}(P_f)$，即可代入求得相對應的常數 a。在這兩種表示法中，常數 a 皆代表某地表加速度值，進而應用在地震易損性評估的失效機率轉換。經由表一的正規化對數常態分佈結果內插得知，95%、50%與 5%信心水準（分別代表 0.05、0.50 與 0.95 失效機率）之正規化常數 u 分別為-1.65、0 與 1.65，可以下列三個方程式表示

$$a = \tilde{a} \cdot e^{-1.65\beta} \quad \text{（95\% confidence）}$$

$$a = \tilde{a} \quad \text{（50\% confidence）}$$

$$a = \tilde{a} \cdot e^{+1.65\beta} \quad \text{（5\% confidence）}$$

其中，標準差β在地震易損性分析中代表採用參數等所隱含的變異性性。如以 95%、信心水準作為基礎，須採用正規化常數 u 分別為-1.65，在同時考慮兩種變異性下，其高信度低損害率（High Confidence Low Probability of Failure, HCLPF）可表示為

$$HCLPF_{50} = A_m \cdot e^{-1.65(\beta_R+\beta_U)}$$

其中

β_R = logarithmic standard deviation for randomness

β_U = logarithmic standard deviation for uncertainty

β_R與β_U代表分析採用參數等所隱含的隨機性（randomness）及不確定性（uncertainty），計算時則可以將其何併考慮為

$$\beta_C = \sqrt{\beta_R^2 + \beta_U^2}$$

依據 EPRI 報告 TR-1019200[8]，修正後較完整之高信度低損害率 HCLPF 應表示為

$$HCLPF_{CDFM} = A_m \cdot e^{-1.65(\beta'_R+\beta'_U)}$$

其中

$$\beta'_R = \sqrt{\beta_R^2 - \beta_{RS}^2}$$

$$\beta_{RS} = 0.2$$

$$\beta'_U = \beta_U$$

未來，將參考此一計算法則推估β'_R與β'_U，以及國內金山核能一廠 1990 年之機率式風險評估結果，進行地震易損度評估驗證工作。

為節省電腦運算時間，如表一的樣本在±3 的區間提供有 13 個位置的失效機率，正規化對數常態累積分佈函數可利用一組 12 階多項式加以近似，如此即可不再進行繁雜的數值積分。其 12 階多項式表示，如下列方程式所示

$$P(u) = a_0 + a_1 u + a_2 u^2 + a_3 u^3 + \cdots + a_{12} u^{12}$$

其中，未知係數為 a_0 至 a_{12}。值得注意的一點，若常數 a 若超過±3 的區間，則此多項式表式法是無效的。配合表一的正規化對數常態累積分佈函數，可獲得 13 條聯立方程式，以及其 13 個未知係數 a_0 至 a_{12} 需要進行求解。再將此聯立方程式簡化成為矩陣表示形式，如下列所示

$$\begin{bmatrix} 1 & u_1 & u_1^{\ 2} & u_1^{\ 3} & \cdots & u_1^{\ 12} \\ 1 & u_2 & u_2^{\ 2} & u_2^{\ 3} & \cdots & u_2^{\ 12} \\ 1 & u_3 & u_3^{\ 2} & u_3^{\ 3} & \cdots & u_3^{\ 12} \\ 1 & u_4 & u_4^{\ 2} & u_4^{\ 3} & \cdots & u_4^{\ 12} \\ \vdots & \vdots & \vdots & \vdots & \ddots & \vdots \\ 1 & u_{13} & u_{13}^{\ 2} & u_{13}^{\ 3} & \cdots & u_{13}^{\ 12} \end{bmatrix} \begin{Bmatrix} a_0 \\ a_1 \\ a_2 \\ a_3 \\ \vdots \\ a_{12} \end{Bmatrix} = \begin{Bmatrix} P(u_1) \\ P(u_2) \\ P(u_3) \\ P(u_4) \\ \vdots \\ P(u_{13}) \end{Bmatrix}$$

最後經由線性代數方法，如 LU 分解法或 SVD 奇異值分解法，皆可有效取得反矩陣，並且與已知向量相乘之後，可求得 12 階多項式之未知係數 a_0 至 a_{12}，如下列所示

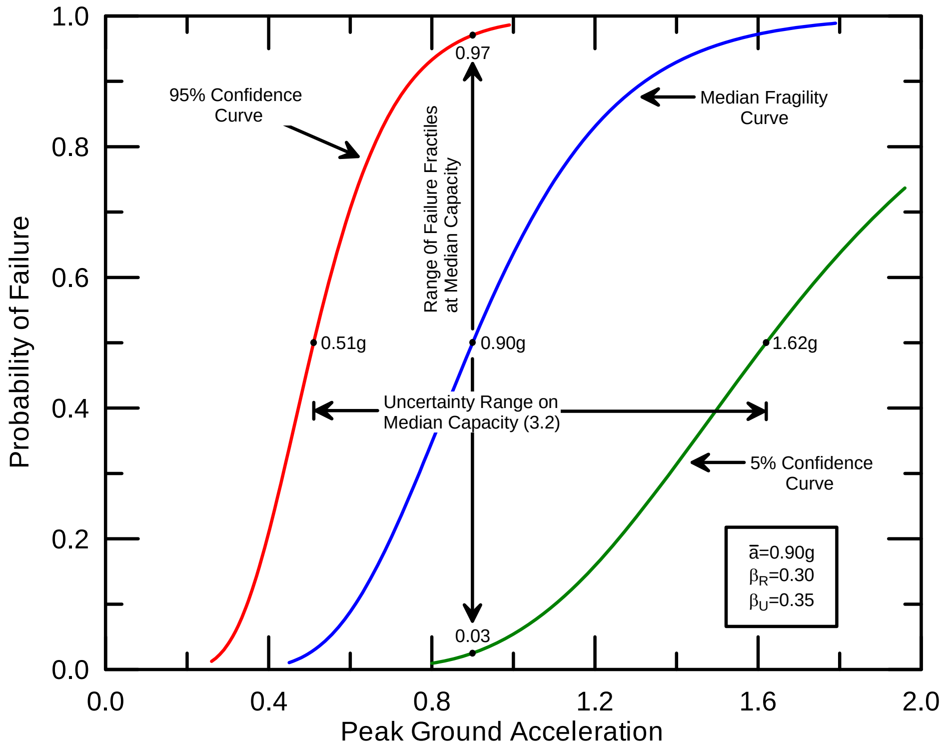
$$\begin{Bmatrix} a_0 \\ a_1 \\ a_2 \\ a_3 \\ \vdots \\ a_{12} \end{Bmatrix} = \begin{bmatrix} 1 & u_1 & u_1^{\ 2} & u_1^{\ 3} & \cdots & u_1^{\ 12} \\ 1 & u_2 & u_2^{\ 2} & u_2^{\ 3} & \cdots & u_2^{\ 12} \\ 1 & u_3 & u_3^{\ 2} & u_3^{\ 3} & \cdots & u_3^{\ 12} \\ 1 & u_4 & u_4^{\ 2} & u_4^{\ 3} & \cdots & u_4^{\ 12} \\ \vdots & \vdots & \vdots & \vdots & \ddots & \vdots \\ 1 & u_{13} & u_{13}^{\ 2} & u_{13}^{\ 3} & \cdots & u_{13}^{\ 12} \end{bmatrix}^{-1} \begin{Bmatrix} P(u_1) \\ P(u_2) \\ P(u_3) \\ P(u_4) \\ \vdots \\ P(u_{13}) \end{Bmatrix}$$

此法可大幅提升運算速率，並準確求得某中位數和平均值之累積分佈機率。為驗證地震易損性曲線建立方法，參考 EPRI 報告 TR-103959 之條件，可分別繪製某一電廠結構（Structure）與設備（Equipment）之地震易損性曲線，如圖六和圖七所示。

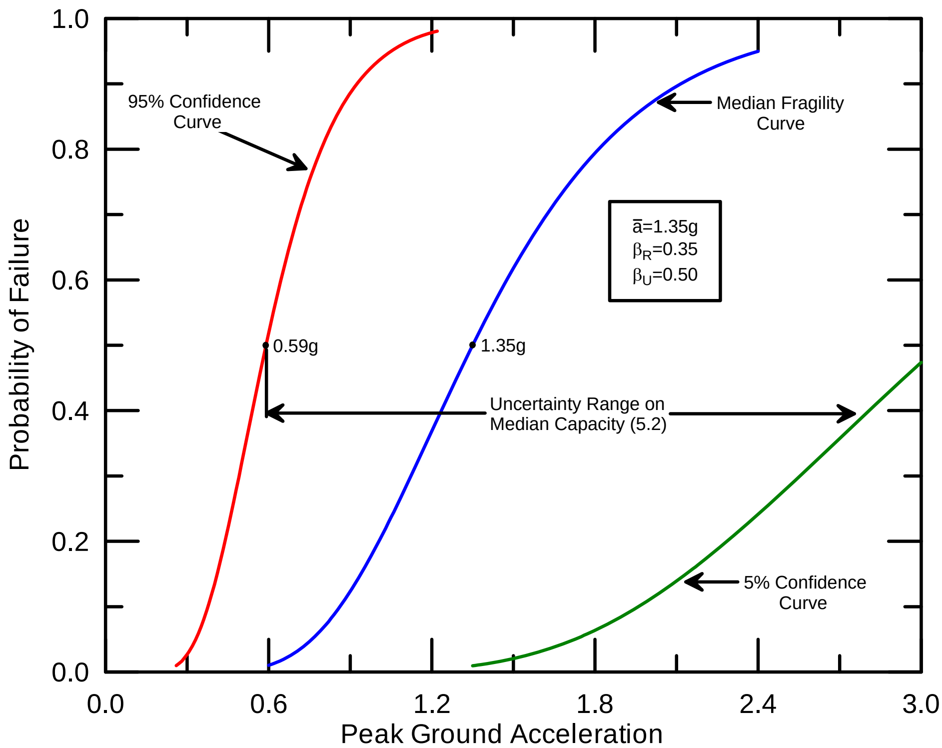
(一) 依據國內金山核能一廠 1990 年之機率式風險評估報告[9]，有關土建結構地震易損度評估所包含的結構物為：

1. 聯合結構廠房（Combination Structure）
2. 緊急泵室（Emergency Pumphouse）
3. 反應爐基座（Reactor Vessel Pedestal）
4. 反應混凝土屏護牆（Reactor Concrete Shield Wall）
5. 抑壓池支撐柱（Torus Support Column）
6. 生物屏護牆（Biological Shield Wall）

其地震易損度評估參數，整理如表二所示



圖六、地震易損性曲線驗證-結構



圖七、地震易損性曲線驗證-設備

表二、核一廠土建結構地震易損度評估參數

Structures	A_m	β_R	β_U	HCLPF	Failure Mode
1. Combination Structure	2.8	0.23	0.34	1.1	Shear wall
2. Emergency Pumphouse	3.8	0.28	0.36	1.3	Shear wall
3. Reactor Vessel Pedestal	5.6	--	--	--	Flexural failure
4. Reactor Concrete Shield Wall	7.5	--	--	--	Flexural failure
5. Torus Support Column	7.8	--	--	--	Machine bolts
6. Biological Shield Wall	8.0	--	--	--	Flexural failure

上列第 3 項至第 6 項結構物座落於聯合結構廠房中，原地震易損度評估結果顯示其可承受之最大地表加速度（Peak Ground Acceleration, PGA）遠大於 3g。另外，第 1 項聯合結構廠房與第 2 項緊急泵室之詳細地震易損度評估參數，則分別整理如表三與表四所示。

表三、核一廠聯合結構廠房地震易損度評估參數

Factors	Median Strength Factor	Standard Deviation for Randomness β_R	Standard Deviation for Uncertainty β_U	Combined Value β_C
Strength	8.14	0.00	0.23	0.23
Inelastic Energy Absorption	1.15	0.02	0.01	0.02
Structural Response	1.00	0.22	0.20	0.30
Modeling	1.00	0.00	0.15	0.15
EQ. Component Combination	1.00	0.05	0.00	0.05
Horizontal EQ. Direction	1.00	0.00	0.00	0.00
Total F. S.	9.3	0.23	0.34	0.41
A_{DBE}= 0.30				
Ground Acceleration Capacity	A_m	β_R	β_U	$HCLPF_{50}$
	2.8	0.23	0.34	1.1

表四、核一廠緊急泵室地震易損度評估參數

Factors	Median Strength Factor	Standard Deviation for Randomness β_R	Standard Deviation for Uncertainty β_U	Combined Value β_C
Strength	3.14	0.00	0.23	0.23
Inelastic Energy Absorption	1.38	0.08	0.06	0.10
Structural Response	2.90	0.26	0.23	0.35
Modeling	1.00	0.00	0.15	0.15
EQ. Component Combination	1.00	0.05	0.00	0.05
Horizontal EQ. Direction	1.00	0.00	0.00	0.00
Total F. S.	12.6	0.28	0.36	0.46
A_{DBE}= 0.30				
Ground Acceleration Capacity	A_m	β_R	β_U	$HCLPF_{50}$
	3.8	0.28	0.36	1.3

（二）依據國內金山核能一廠 1990 年之機率式風險評估報告，有關設備結構地震易損度評估所包含的結構物為：

1. Condensate Storage Tank
2. Diesel Oil Storage Tank
3. Reactor Vessel Stabilizer
4. Reactor Vessel Support Skirt
5. Core Support Structure
6. Fuel Assembly
7. Air Accumulator for SRV and MSIV
8. Recirculation Pumps/Motors
9. Main Control Boards
10. Relay Logic Panels
11. Control Room Ceiling
12. Essential Service Water System HCC

其地震易損度評估參數，整理如表五所示

表五、核一廠設備結構地震易損度評估參數

Equipments	A_m	β_R	β_U	HCLPF	Failure Mode
1. Condensate Storage Tank	1.15	0.51	0.33	0.29	Shell buckling
2. Diesel Oil Storage Tank	2.10	0.39	0.35	0.62	Shell buckling
3. Reactor Vessel Stabilizer	5.55	0.27	0.36	1.96	Stabilizer bracket
4. Reactor Vessel Support Skirt	1.22	0.27	0.36	0.43	Vessel to girder bolts
5. Core Support Structure	1.05	0.40	0.38	0.29	Shroud support leg buckling
6. Fuel Assembly	1.70	0.34	0.33	0.56	Collapse of fuel rods
7. Air Accumulator for SRV	0.79	0.37	0.48	0.19	Anchorage failure
8. Recirculation Pumps/Motors	1.77	0.32	0.35	0.59	Snubber failure
9. Main Control Boards	2.65	0.35	0.38	0.79	Functional
10. Relay Logic Panels	2.65	0.35	0.38	0.79	Functional
11. Control Room Ceiling	0.35	0.32	0.40	0.11	Celting component failure
12. Essential Service Water System	2.65	0.31	0.44	0.77	Functional

第 1 項冷凝水貯存槽與第 2 項柴油貯存槽之詳細地震易損度評估參數，則分別整理如表六與表七所示。

表六、核一廠冷凝水貯存槽地震易損度評估參數

Factors	Median Strength Factor	Standard Deviation for Randomness β_R	Standard Deviation for Uncertainty β_U	Combined Value β_C
Strength	3.84	0.00	0.21	0.21
Inelastic Energy Absorption	1.00	0.00	0.00	0.00
Spectral Shape	1.00	0.50	0.00	0.50
Damping	1.00	0.00	0.00	0.00
Modeling	1.00	0.00	0.16	0.16
Modal Combination	1.00	0.08	0.00	0.08
EQ. Component Combination	1.00	0.05	0.00	0.05
Soil Structure Interaction	1.00	0.00	0.20	0.20
Horizontal EQ. Direction	1.00	0.00	0.00	0.00
Total F. S.	3.8	0.51	0.33	0.61
A_{DBE}= 0.30				
Ground Acceleration Capacity	A_m	β_R	β_U	$HCLPF_{50}$
	1.2	0.51	0.33	0.3

表七、核一廠柴油貯存槽地震易損度評估參數

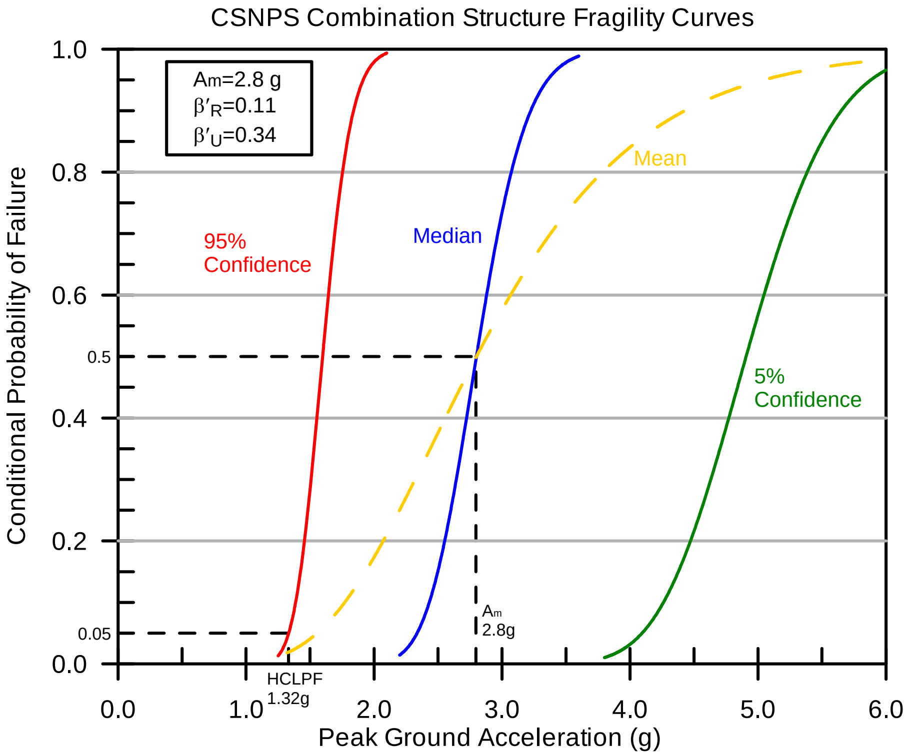
Factors	Median Strength Factor	Standard Deviation for Randomness β_R	Standard Deviation for Uncertainty β_U	Combined Value β_C
Strength	6.90	0.00	0.21	0.21
Inelastic Energy Absorption	1.00	0.00	0.00	0.00
Spectral Shape	1.00	0.39	0.00	0.39
Damping	1.00	0.00	0.00	0.00
Modeling	1.00	0.00	0.19	0.19
Modal Combination	1.00	0.05	0.00	0.05
EQ. Component Combination	1.00	0.00	0.00	0.00
Soil Structure Interaction	1.00	0.00	0.20	0.20
Horizontal EQ. Direction	1.00	0.00	0.00	0.00
Total F. S.	6.9	0.39	0.35	0.52
A_{DBE}= 0.30				
Ground Acceleration Capacity	A_m	β_R	β_U	$HCLPF_{50}$
	2.1	0.39	0.35	0.6

表八、核一廠土建結構耐震餘裕更新

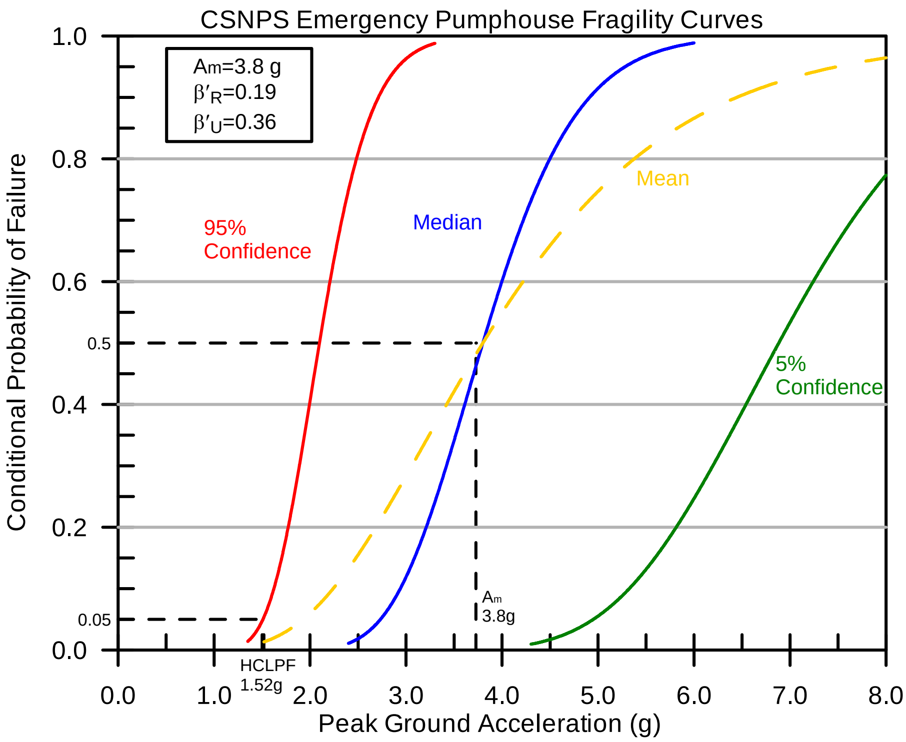
Structures	A_m	β_R	β_U	$HCLPF_{50}$	β'_R	$HCLPF_{CDFM}$
1. Combination Structure	2.80	0.23	0.34	1.10	0.11	1.32
2. Emergency Pumphouse	3.77	0.28	0.36	1.31	0.19	1.52
3. Reactor Vessel Pedestal	5.60	--	--	--	--	--
4. Reactor Concrete Shield Wall	7.50	--	--	--	--	--
5. Torus Support Column	7.80	--	--	--	--	--
6. Biological Shield Wall	8.00	--	--	--	--	--

表九、核一廠設備結構耐震餘裕更新

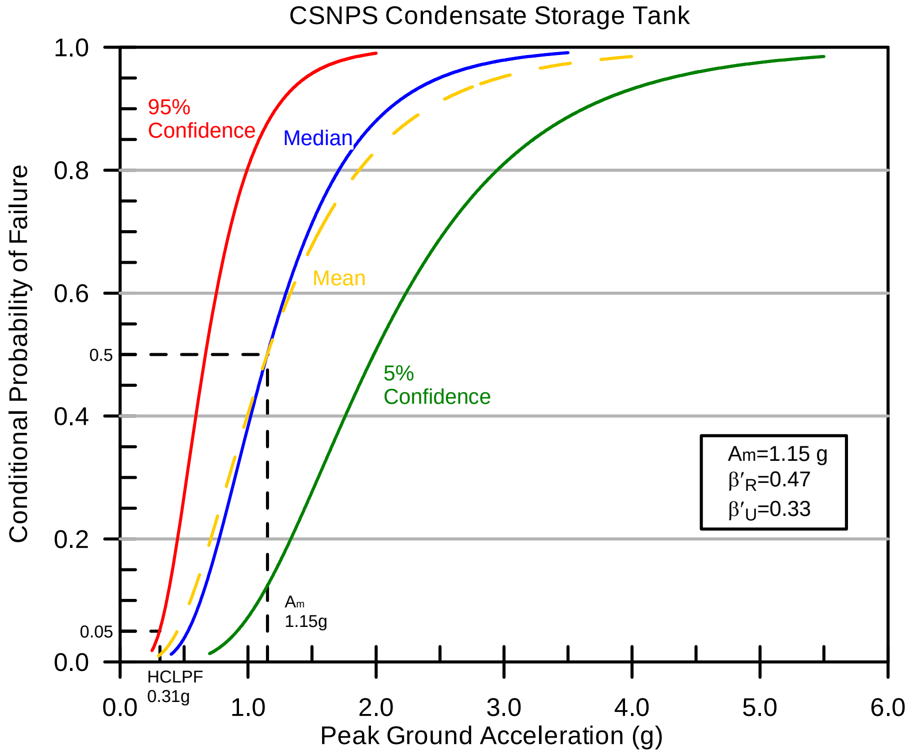
Equipments	A_m	β_R	β_U	$HCLPF_{50}$	β'_R	$HCLPF_{CDFM}$
1. Condensate Storage Tank	1.15	0.51	0.33	0.29	0.47	0.31
2. Diesel Oil Storage Tank	2.07	0.39	0.35	0.61	0.34	0.67
3. Reactor Vessel Stabilizer	5.55	0.27	0.36	1.96	0.18	2.27
4. Reactor Vessel Support Skirt	1.22	0.27	0.36	0.43	0.18	0.50
5. Core Support Structure	1.05	0.40	0.38	0.29	0.35	0.32
6. Fuel Assembly	1.70	0.34	0.33	0.56	0.27	0.63
7. Air Accumulator for SRV	0.79	0.37	0.48	0.19	0.31	0.21
8. Recirculation Pumps/Motors	1.77	0.32	0.35	0.59	0.25	0.66
9. Main Control Boards	2.65	0.35	0.38	0.79	0.29	0.88
10. Relay Logic Panels	2.65	0.35	0.38	0.79	0.29	0.88
11. Control Room Ceiling	0.35	0.32	0.40	0.11	0.25	0.12
12. Essential Service Water System	2.65	0.31	0.44	0.77	0.24	0.87



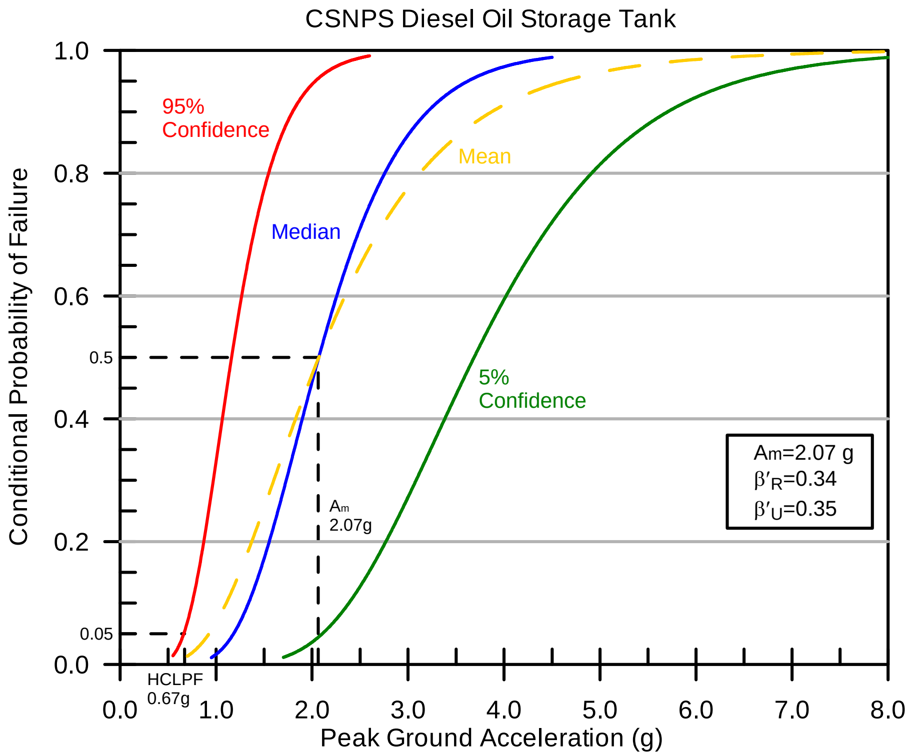
圖八、核一廠聯合結構廠房之地震易損性曲線



圖九、核一廠緊急泵室之地震易損性曲線



圖十、核一廠冷凝水貯存槽之地震易損性曲線



圖十一、核一廠柴油貯存槽之地震易損性曲線

<u>12 月 9 日，星期五 至 12 月 10 日，星期六</u>

12 月 9 日清晨由 S&A 公司 Mr. Mo-Hwa Wang 親自開車，前往波士頓羅根國際機場搭乘美國航空 AA 25 班機，飛往洛杉磯國際機場，最後搭乘中華航空 CI 5 班機飛返桃園機場，因國際換日及長途飛行，於 12 月 10 日抵達。

三、 心得

此行前往美國史蒂文生公司(Stevenson & Associate, S&A)，研習核電廠結構地震動態分析與安全審查技術，不僅有關專業技術之學習，亦包括對新事物之見識與人際關係之拓展等。以下分別就進行工作提出心得報告：

(一) 本次實習機會學習到許多解決問題之思考邏輯與應用方式，例如應用 Fortran 數值模擬程式撰寫機率密度函數與地震易損性曲線，有助於解決許多複雜之計算，並徹底了解方程式本身的內涵，係為一項重要的訓練，而所建構之計算程式亦可應用於核能電廠安全評估之工具。

(二) 最主要收穫便是了解及增進核電廠耐震餘裕安全評估之技術及經驗，以客觀之耐震餘裕分析技術，評估核電廠安全餘裕。相信對提昇我國核能電廠之安全分析技術有很大之幫助。

(三) 實習期間工作與交付之任務繁多，學習內容緊湊。S&A 公司亦毫無保留地指導職相關之專業知識。因此，雖然辛苦但也習得很多新的概念，並徹底地了解電廠耐震餘裕安全評估之考量條件與相關作法與工程應用，以作為我國評估該設施安全性之重要參考。

(四) 研習核電廠耐震餘裕安全評估之分析架構與執行程序，提供適用於我國核能電廠之耐震餘裕安全評估，作為我國相關技術之重要基礎。

(五) 利用此次實習機會，體驗了美國的生活與風土民情。並利用在 S&A 公司上班之機會，與資深工程師 Mr. Mo-Hwa Wang、Mr. John Holland、Mr. Joshua Hart 等成為摯友，並結識了 S&A 公司總裁 Mr. Walter Djordjevic、副總裁 Dr. T. M. Tseng、專案經理 Dr. Keith Xu、資深工程師 Dr. Yi-Lun Chu 與 Mr. Sung-June Kim 等公司高層。對於個人之人際關係拓展，不啻為另一重要收穫。

四、 建議事項

本次的公差行程，在相關領域的學者進行研究方向的討論，確實可以體會到隨著科技的進步，研究方向已不再藉由合理簡化的模式進行，取而代之的，是結合更多的參數進行實際的模擬。未來，可能可藉由雙方交流的機會，或許可將分析問題的擬真與評估獲得更佳的成果。

因應核能發電復甦和全球化核能工業的需求，各主要核能工業國家近來已積極投入更多人力與經費，期望能夠以更有效率更經濟的發電方式，降低目前全球嚴重暖化的速度。建議我國亦應密切注意國際趨勢，鼓勵學術單位和產業界的核能研究開發與產能，培養更多未來願意參與的學生，提供獎學金和就業機會。另一方面，加強國際合作以建立彼此的交流管道，互相交換人才與研究成果，以瞭解最新的資訊發展，進而掌握正確的發展方向與契機。

S&A 公司管理方式自由，每位員工皆有自己辦公室，且上下班時間不須刷卡管制，時間自主，對所屬工作負責完成後即可，於時間運用上有較大彈性，更易培養對工作任務之歸屬感與負責任之態度。然國情不同，此方式是否適合國內，則還須更進一步之研究與探討。

附件一

史蒂文生公司**(Stevenson & Associate, S&A)**簡介

Stevenson & Associates

CAPABILITIES & SERVICES OVERVIEW

Stevenson & Associates (S&A) has provided expert structural and mechanical engineering consulting services to the commercial nuclear power industry since 1981. We are internationally recognized for our extreme external hazard consulting, particularly seismic analysis and design. While these services are premier areas of expertise for our company, S&A also provides a wide range of structural and mechanical engineering consulting services.

Our capabilities and Services Include:

- **Design Reviews and Audits**
- **Analysis and Design of Structural Steel, Reinforced Concrete and Masonry**
- **Piping Analysis and Qualification**
- **Experienced Based Piping Qualification**
- **Piping System Sizing, Layout and Design**
- **Buried Piping Design and Analysis**
- **Snubber Reduction Programs**
- **Failure Analysis: Fracture Mechanics and Fatigue**
- **On-Site Field Engineering**
- **Systematic Evaluation and Upgrade programs**
- **Safety and Operability Evaluations**
- **Vibration Monitoring and Evaluation**
- **Extreme Load Design**
- **Equipment Qualification: Seismic and Environmental**
- **Seismic and External Event Probabilistic Risk Assessment (PRA)**
- **Services for Design Basis Threat (DBT)**

S&A is headquartered in the Boston area with full service offices in Chicago and Cleveland, and project offices in the greater Minneapolis and Hartford (CT) areas. S&A has over forty experienced engineers with many possessing advanced graduate degrees from top technical universities. Over one-half of the engineering staff has professional engineer registrations. S&A also collaborates with well-known professors at leading academic institutions such as MIT, Stanford and the University of Illinois.

Our clients include most of the US utilities that operate nuclear power plants, architect-engineering companies, US Department of Energy, US Military, and the USNRC. S&A also supports nuclear industry groups such as owners groups and task forces, the Nuclear Energy Institute (NEI) and the Electric Power Research Institute (EPRI). S&A's international clients include China, Korea, Taiwan, Argentina, Canada, Hungary, Czech Republic, Russia, Holland, Switzerland, and South Africa. S&A has provided services to international agencies such as AECL, CNSB and IAEA.

S&A administers an Appendix B (10CFR50) Quality Assurance (QA) program for safety-related work which is audited and certified by NUPIC. S&A has been performing QA safety-related work for the nuclear industry since its inception in 1981.

www.vecsa.com • Boston • Chicago • Cleveland • Minneapolis

Stevenson & Associates

CIVIL-STRUCTURAL ENGINEERING SERVICES

Stevenson & Associates (S&A) has extensive experience in the evaluation of buildings, structures, and mechanical and electrical systems for the effects of extreme loads resulting from both natural and manmade phenomena. In addition, S&A has Project Management Capability to build dedicated teams to develop the complete design/modification package from design through implementation. Our skill set includes the analysis and design of steel, concrete and masonry structures.

The S&A Civil-Structural Services include but are not limited to the following:

- **Extreme Load Design (Seismic, Tornado, Wind/Hurricane, Blast)**
- **Flood**
- **Impact Analysis**
- **Explosions**
- **Soil-Structure Interaction (SSI) Analyses**
- **Seismic Fragility and Probabilistic Risk Analysis – Seismic Margin & SPRA**
- **Structural Analysis – Linear & Nonlinear**
- **Analysis and Design of Structural Steel, Reinforced Concrete and Masonry**
- **Finite Element Analysis Capabilities**
- **Equipment Qualification**
- **Expert Consultation**

Extreme Load Design
S&A is a world renowned consulting firm in analysis and design for natural phenomena hazard for structures, systems and components (SSCs). Services range from analyzing and designing SSCs for seismic, wind and pressure loadings to developing computer models and performing dynamic response analyses.

Soil-Structure Interaction Analysis
S&A has been in the forefront of SSI analysis for over 25 years. S&A has conducted safety-related analyses for numerous clients and for design basis applications using the SASSI™ and EKSSI™ codes. S&A maintains and markets these codes throughout the world for resale with QA validation and verification. S&A's SuperSASSI™ code is the first commercially developed PC-based SSI code developed for the general market.

Seismic and External Hazard Probabilistic Risk Assessment (PRA)
S&A is a highly experienced consultant in the field of seismic and external hazard PRA. S&A remains active in this field providing these services around the world on an ongoing basis. S&A has performed more seismic PRA's and teaching seminars than any firm in the US and maintains the same staff that performed the work for IPEEE in the 1990's. S&A has also performed many Seismic Margin assessments in compliance with IPEEE requirements.

Steel and Concrete Design

S&A provides structural design services to numerous clients with complete capabilities ranging from plant modifications to complete building design services and professionally sealed design drawings to conform to national and state codes.

Linear/Non-linear Structural Analysis and Finite Element Method (FEM) Modeling

S&A routinely performs all manner of structural and FEM modeling for complex structures, systems and components. Evaluations involve both static and dynamic analysis for both linear and non-linear systems. Loadings involving pressure transients, blast, natural phenomena and fluid loads are just some of the engineering problems S&A has addressed. S&A maintains a stable of QA verified and validated computer codes for such analysis purposes including ANSYS™, ADINA™, GTSTRUDL™, PDSTRUDL™, SAP2000™ and many of its own codes developed in-house such as SPECTRA™ and EDASP™.

Equipment Qualification

S&A provides qualification services for all types of mechanical and electrical equipment. S&A is conversant with ASME Appendix N, and IEEE 323 and 344 Standard requirements and maintains membership (and chairmanship for App. N) in these same committees. S&A performs detailed stress analysis for components including stress reports, modeling, in-situ modal testing for frequencies and mode shapes, and anchorage analysis and design.

Expert Consultation

S&A provides expert consulting services to clients in need of counsel and as representatives when meeting with reviewing and regulatory agencies.

Stevenson & Associates

PIPING ANALYSIS CAPABILITIES

Stevenson & Associates (S&A) has extensive experience in all aspects of Piping System Design, Analysis, Qualification and Field Implementation. The S&A Staff specializes in providing expert consulting services and engineering capabilities for complex design issues. In addition, this staff has Project Management Capability to build dedicated teams to handle the complete package from design thru implementation.

- **ASME Piping Design & Analysis**
- **ASME Boiler and Pressure Vessel Code Consulting**
- **Component Support Design and Analysis**
- **Mechanical Finite Element Analysis Capabilities**
- **Vibration Testing, Monitoring, Analysis & Design – PCMODAL, PCODS, PCMONITOR**

ASME Piping Design and Analysis
S&A performs ASME Class 1, 2 and 3 Code compliance piping analysis and design for many nuclear power plants in the US. The services range from modeling, thermal, mechanical (water hammer), pressure, seismic and other dynamic loadings to full certified design including certified design specifications and design stress reports. S&A is also fully fluent in ANSI B31.1; B31.3 & B31.7 design code requirements. S&A maintains numerous QA verified and validated computer codes including PIPESTRESS™, ADLPIPE™, CAEPIPE™, PDSTRUDL™ and its own GUIPIPE™.

ASME Boiler & Pressure Vessel Code Consulting
S&A provides expert ASME code consultation for Section III, Section VIII and Section XI compliance. S&A has analyzed and designed piping, valves, pumps, heat exchangers, tanks, vessels and other pressure-retaining components to code requirements. S&A's senior engineers have certified qualifications in ASME N626.3 and Appendix XXIII. S&A is also expert in Class CC and MC code compliance. S&A personnel participate and sit on numerous ASME Code Committees including the main oversight committee for Section III.

Mechanical Finite Element Method (FEM) Evaluation
S&A performs all manner of FEM modeling for mechanical components, subcomponents and fixtures. Evaluations involve both static and dynamic analysis for both linear and non-linear systems. Loadings involving pressure transients, thermal transient and dynamic including seismic and fluid loads. S&A maintains a stable of QA verified and validated computer codes for such analysis purposes including ANSYS™, ADINA™, GTSTRUDL™, PDSTRUDL™, SAP2000™ and many of its own codes developed in-house such as SPECTRA™ and EDASP™.

Vibration Testing and Monitoring
S&A maintains a mobile laboratory of testing equipment for purposes of obtaining modal, forced response and ambient signature vibration levels. S&A performs low impedance, modal testing for the purposes of establishing characteristic frequencies and mode shapes, operating deflected shapes for rotating equipment, and vibration monitoring setups for measuring ambient and problematic vibration levels in equipment and piping. S&A provides analysis services and remedial designs using its own codes: PCMODAL, PCODS and PCMONITOR.

www.vecsa.com • Boston • Chicago • Cleveland • Minneapolis

Stevenson & Associates

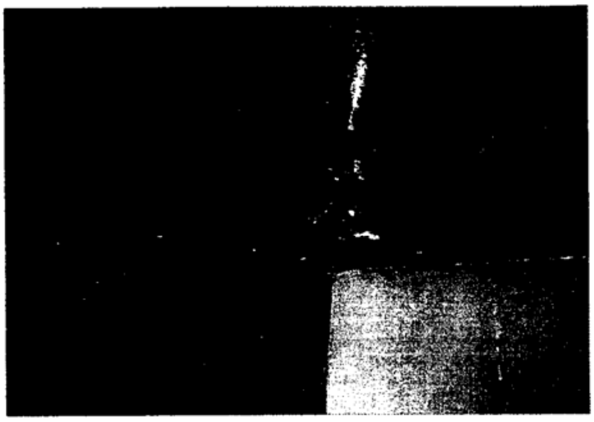
PIPING SYSTEM VIBRATION MONITORING & REMEDIAL DESIGN

Stevenson & Associates (S&A) has been providing vibration measurement, analysis and design services since 1983 for numerous utilities and agencies. Services include modal testing and analysis, operating deflected shape measurement and analysis, ambient vibration determination (measurement), and long-term (semi-permanent) vibration monitoring for both in-containment and balance-of-plant piping and equipment systems. S&A also provides detailed analysis and remedial design services to mitigate and reduce vibrations.

S&A developed a methodology that addressed unacceptably high, flow-induced vibration levels for a main steam line outside of containment. The vibrations and associated loading have resulted in damaged supports, support hardware, support anchorages and insulation during plant operation.

S&A monitored the vibration levels in an attempt to quantify the level and frequency content of the operating vibrations. A new design concept using vibration dampers manufactured by GERB was employed for the main steam lines. The existing lateral and longitudinal restraints including snubbers were removed in favor of GERB dampers. New, nonlinear piping models were developed to achieve the desired reduction in vibration levels and yet meet the loading combination requirements including dynamic loads (seismic and water hammer).

GERB dampers Installed



Excessive Loads
caused this Support to Fracture

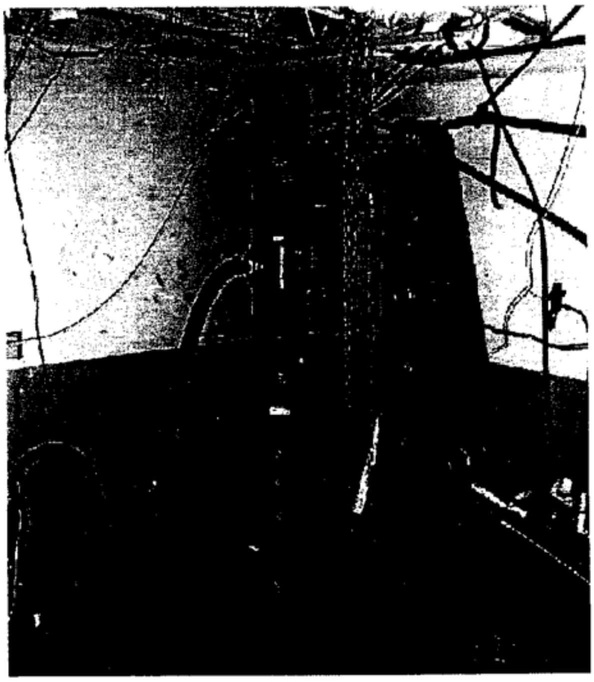
BURIED HIGH DENSITY POLYETHYLENE (HDPE) PIPING

Development of Design Methodology

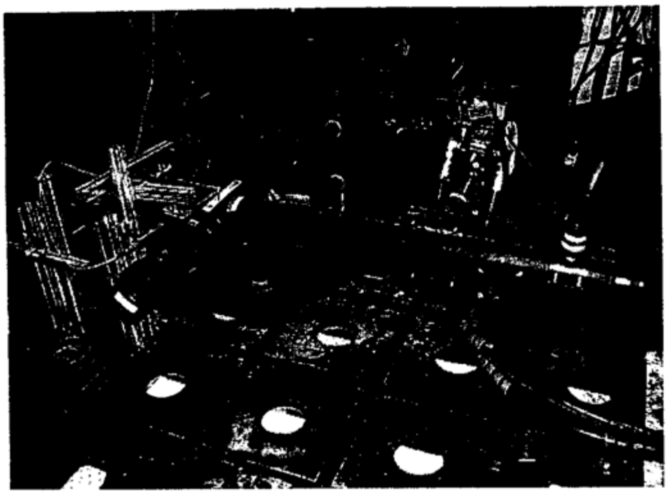
As principal investigator for this EPRI (Electric Power Research Institute) sponsored project, Stevenson & Associates (S&A) have been pioneers in developing the methodology, design and licensing of High Density Polyethylene (HDPE) piping to replace corroded and degraded buried low carbon steel ASME Class 2 and 3 piping systems in commercial Light Water Reactors operating within the United States. Due to the advantageous cost and durability of (as demonstrated in other commercial industries), it was concluded that ASME code inclusion of HDPE piping was logical. Historically the ASME Code has not actively supported non-metallic piping in power plants. However, it has been successfully used in commercial applications such as water mains and natural gas pipelines. S&A developed a methodology that addressed all applicable failure modes including allowable stress and strain limits and design criteria for HDPE pipe in nuclear power plants. The methods included comply with ASME Power Piping Code, B31.1-2004 and Section III of the ASME Boiler and Pressure Vessel Code. Extensive use was made of industrial research, data and experience over 40 years of use of high-density polyethylene piping. Allowable stresses are based on data published in these sources for Design and Service Levels A-D.

HDPE Pipe Testing Program

S&A also designed and implemented a testing program of several HDPE components to supplement the body of data available for this piping. S&A selected the laboratory performing the testing for tensile strength, fatigue-induced stress, and developed stress indices as a result of these tests. In addition, S&A commercially dedicated a vendor to be an acceptable material supplier to supply test materials and specimens for this test program. S&A developed a test plan for tensile testing and fatigue testing under a formal quality assurance program compatible with ISO-9000. The test plan met the requirements of ASME III Appendix XXIII and included a detailed description of the each of the testing tasks, detailed definition of data recoding tasks and the associated data recording sheets, definition of the data reduction to be conducted, the method of results presentation, and development and verification and validation of any computer software to be used in the data reduction process.



HDPE Pipe Testing Program



HDPE Pipe Testing Program

The testing included the development of full range stress-strain properties of the HDPE piping at four temperatures for both new and thermally aged conditions. The testing also included fatigue testing of fusion butt welded HDPE pipe and 5 segment miter elbow joints. The data obtained was used to determine the frequency of the cycling effect on fatigue life and Code acceptable piping stress intensification factors (SIFs). Both an ambient and elevated temperature set of tests were run.

The results of the tests were used to identify the controlling frequency for cyclic fatigue. With this data S&A developed Code acceptable piping stress intensification factors (SIFs) for fusion butt joints in accordance with the guidance of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Appendix II, Article II - 2000 and created the S-N curve that can be used for design in the ASME BPVC, Section III, Division 1.

Implementation of Replacement Piping at a Nuclear Power Plant

S&A implemented the methodology by authoring "ASME Code Case N-755, High Density Polyethylene (HPDE) Buried Pipe, Section III, Division I, Class 3", Revision 0, dated March 22, 2007. Using this code case S&A conducted the detailed design calculations for the "first-of-its-kind" replacement of eight buried carbon steel piping systems for Duke Power's Catawba Nuclear Station. These designs are currently scheduled for installation at the site in the near future.

Stevenson & Associates

HIGH ENERGY LINE BREAK (HELB)

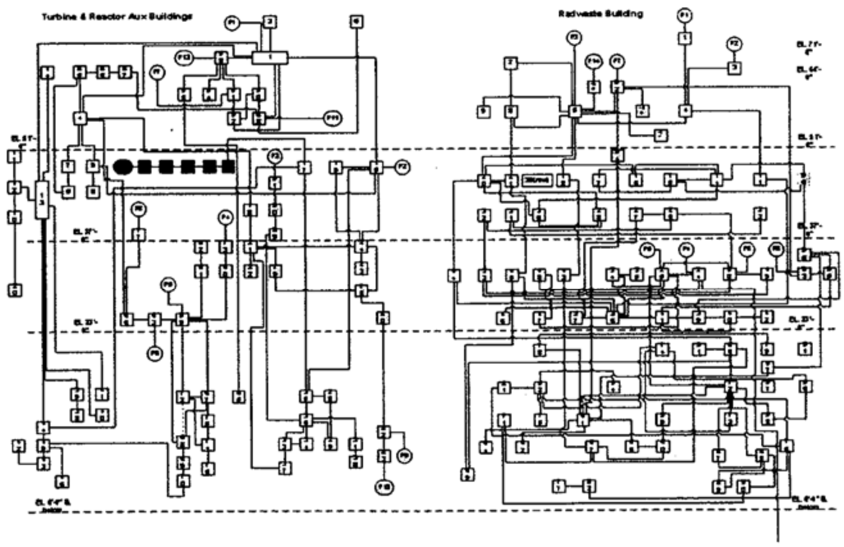
Stevenson & Associates (S&A) has performed complete engineering designs for nuclear plant sites to deter and withstand High Energy Line Breaks (HELB) that have been imposed by NRC Letter to Nuclear Power Plant Licensees dated December 1972 (a.k.a. Giambusso Letter), SRP sections 3.6.1 and 3.6.2, and/or ANSI 58.2. As such, S&A has participated in conceptual designs, performed walk downs to determine vulnerabilities, and executed designs to withstand postulated effects of a line break such that the plant can be operated or shut down safely.

S&A has performed complete assessments the effects of line breaks include: pressurization of compartments; temperature gradients; humidity; jet forces; pipe whipping; missiles; internal flooding; etc. The protection from line breaks include: capacities of structures; pipe supports; pipe whip restraints; protection of plant equipment, maintenance of HELB boundaries or barriers; equipment environmental qualification; internal flooding control; etc.

Consulting services include:

- **Performance of plant walk downs.**
- **Determination of HELB paths and compartment volumes.**
- **Analysis of piping systems.**
- **Structural analysis of HELB boundaries.**
- **Evaluation of structures/components for jet forces or pipe whip effects.**
- **Evaluation/Design of pipe supports and/or pipe whip restraints.**
- **Development of HELB boundary control and surveillance procedures.**
- **GOTHIC ™ Analysis for compartment & sub-compartment response to high energy line breaks.**
- **RELAP ™ Analysis for transient simulations during postulated accidents.**
- **TRACE ™ Analysis for 3D Thermal-Hydraulic accident assessments.**

Compartment pressure
flow model



www.vecsa.com · Boston · Chicago · Cleveland · Minneapolis

Stevenson & Associates

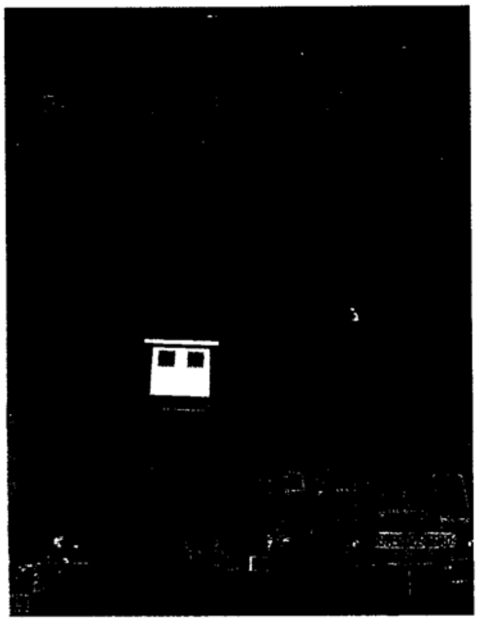
SERVICES FOR DESIGN BASIS THREAT (DBT)

Stevenson & Associates (S&A) has performed complete engineering designs for nuclear plant sites to deter and withstand design basis threats (DBT) that have been imposed by 10CFR Part 73. As such, S&A has performed walkdowns to determine vulnerabilities, participated in conceptual designs, and executed designs to withstand postulated malevolent acts of terrorism against nuclear power plants.

Areas of expertise include detailed blast analysis, evaluation of facilities and buried utilities to blast effects, design of vehicle barriers and gates, missile impact, and design of blast and bullet-resistant facilities and enclosures. S&A has performed complete site DBT engineering projects from the conceptual design phase through on-site construction management of the new installations. S&A has implemented these DBT designs for more than 10 nuclear plant sites.

Consulting services include:

- **Blast Evaluations (TNT, ANFO, H_2)**
- **Determination of Minimum Safe Standoff Distance (MMSD)**
- **Vehicle Impact, Barge (Ship) Impact**
- **Intake Structure Hardening (waterfront defenses)**
- **CAS Hardening**
- **Facility Hardening**
- **Gate and Barrier Design**
- **Tower Design**
- **Ballistic-Resistant Enclosure Design**
- **Cratering Analysis**
- **B5B Evaluations**
- **Communication, Alarm, Detection, Surveillance Systems**
- **Construction Management**
- **Perimeter Design, Delay Obstacles**



Ballistic-Resistant Enclosure being installed

www.vecsa.com • Boston • Chicago • Cleveland • Minneapolis

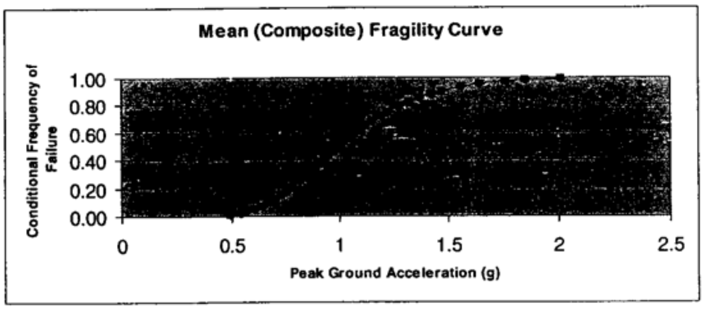
SEISMIC & EXTERNAL EVENT PROBABILISTIC RISK ASSESSMENT (PRA)

Stevenson & Associates (S&A) has provided seismic and external event Level 1 and 2 PRA services for nuclear power plant assessment of core damage frequency and large early release frequency since 1991. The majority of seismic PRAs performed in the 1990s for compliance with the requirements of IPEEE (Individual Plant External Event Examination) and NUREG-1407 are performed by S&A. S&A has been involved in numerous international SPRAs as well. S&A has provided SPRA consulting services for FPL, Entergy, OPPD, Xcel, and Dominion plants, and the Department of Energy and the US Military among others.

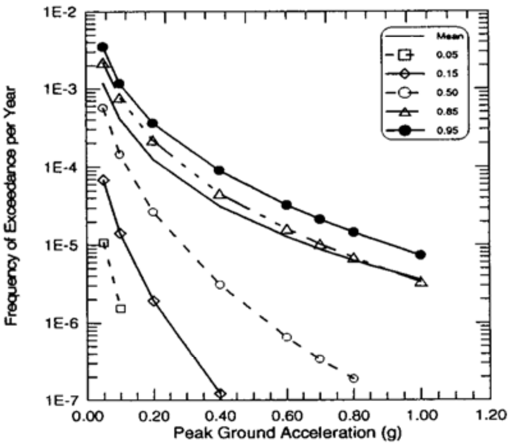
S&A provides the following services in support of SPRAs:

- **Walkdowns to categorize, screen and collect engineering information in support of fragility determinations**
- **Calculations of equipment fragilities using Separation of Variables (SV) and Conservative Deterministic Failure Method (CDFM) approaches**
- **Development of seismic (and external event wind) fragilities including CDFM (HCLPF) and median capacities and associated variability's**
- **Quantification of seismic risk to determine Frequencies of Core Damage (CDF) and Large Early Release (LERF) using in-house and commercially available software tools**
- **Determination of major contributors to seismic risk at the system and component level**

S&A is also performs numerous Seismic Margin studies for its clients in accordance with EPRI Report NP-6041. S&A has performed Seismic Margins studies for Exelon, Entergy, FPL, NMC, First Energy, NPPD and Constellation plants among others.



Seismic Fragility Curve



Seismic Hazard Curves