審查意見摘要表

(第十七章 ~ 附錄E)

第十七章　品質保證

編號	內　　　　容
17-01	1.台電、奇異及石偉公司品保方案之執行次序及整合方式
	2.奇異公司下包商之品保方案澄清
17-02	可靠性品保方案不符 NRC 要求
17-03	1.說明核發處權責有關非破壞檢測之執行
	2.工地品保小組何時成立
17-04	1.Quality Class 中 R 的系統或組件為何，請說明
	2.Quality Class 之 G 組件之規範為何
17-05	1.核工處與品質處之權責相同，請明確訂定
	2.請補充核安會和核後端處之權責
	3.請說明「全面品質管理」之政策聲明下品質處參與程度
	4.請說明全面品質管理的依據標準
17-06	1.請澄清主承包商之下包商所適用品保方案控管問題
	2.請增列核後端處之權責
17-07	請說明建廠期間之品質管制執行方案
17-08	請說明核能同級品之執行方案
17-09	請提供龍門計劃程序書之目錄及時程
17-10	請明確界定品保方案引用之法規及標準之版次
17-11	1.請提供奇異公司龍門計劃之總部及駐廠組織表
	2.請增列品保經理有關「停工要求」之職權
	3.請奇異公司在品保方案上正式簽署
17-12	1.請補充文件管制範圍之內容
	2.請增列「建立各類文件一總覽表」之要求
17-13	1.請澄清採購管制介面之權責歸屬
	2.請補充 QA 單位應對供應商相關作業作驗証
	3.請增訂採購管制須符合 R.G.1.38 要求
17-14	1.請澄清品保方案是否作環境條件的考量
17-15	請澄清品保方案第八章到第十三章僅訂定施工現場的品保管制，未提及原供應廠商部份

編號	內　　　　容
17-16	1.請增充「檢驗點經指定檢驗員驗証」之規定
	2.請增充必要的停留檢驗點之檢証規定
17-17	請澄清對安全有關之作業，如有更動需有原審核之同等級管制
17-18	1.請澄清是否有「應作品質趨勢之分析，重大結論提報高層主管」之品保作業
	2.請澄清品質處在器材管制上之權責
17-19	請澄清「品保單位書面認同改正行動適切性」之要求
17-20	請澄清是否品保記錄儲存設施符合 R.G.1.88 之規定
17-21	1.請澄清是否符合 SRP 17.1 II.18A4「稽查經常被忽略的部份」。
	2.請澄清品保單位是否對稽查數據作分析，並陳報高層審核。

第十八章　人因工程

編號	內　　　容
18-01	控制盤面及資訊展示應考量配合度並包含輔助系統
18-02	1.國人之人因應納入考量，運轉員之介面應定義清楚
	2.現場閥位指示應符合規範要求
	3.主控制盤應有參考書面資料之位置
18-03	1.主控制室設計與傳統型之差異
	2.細部設計時應完成使用者手冊
	3.澄清軟體錯誤時之自動偵測功能
	4.國人之人體工學資料納入考量
	5.操作人員應變能力之確保
18-04	人因工程之適用範圍及設計小組成員資格
18-05	正常運轉時應維持適當之工作負荷

第十九章　嚴重核子事故分析

編號	內　　容
19-001	1.圖表標示不清楚之澄清
	2.電廠全黑(SBO)事故分析序列之澄清
19-002	1.嚴重核子事故造成圍阻體最高壓力之澄清
	2.高壓熔融排放事故(HPME)造成圍阻體失效分析之澄清
19-003	1.氫氣燃燒之假設及分析結果
	2.除壓力分析結果外而提供各項參數之變化圖
19-004	1.MAAP分析程式版本之差異性
	2.嚴重核子事故實驗結果之引用
19-005	1.破裂閥受到溫度變化之影響
	2.圍阻體過壓保護系統(COPS)設計容量
19-006	1.FCI嚴重核子事故實驗之引用
	2.FCI嚴重核子事故實驗之引用
	3.燃料與水作用(FCI)產生蒸氣爆炸之機率
19-007	1.集水池保護裝置通道數目之設計問題
	2.集水池保護裝置通道寬度之設計
19-008	1.MAAP程式之使用版本
19-009	1.嚴重核子事故處理之策略
	2.爐外冷卻以預防反應爐破裂之策略
	3.未來嚴重事故處理計劃(SAM)使用之程式
19-010	1.圍阻體內襯洩漏問題
19-011	1.圍阻體因應氫氣燃燒之設計
	2.當圍阻體沒有惰化時之影響為何
	3.圍阻體衝充氮時間之澄清
	4.被動式自動觸發結合器使用之可能性
19-012	1.COPS設定點之選擇
19-013	1.圍阻體旁通模式之應用
19-014	1.圍阻體耐壓的問題
19-015	1.補充說明核四廠圍阻體之重要特性
	2.圍阻體設計總結之補強(文字上)
19-016	1.核四廠因應嚴重核子事故設計特性之簡述

編號	內　　　容
19-017	1.是否使用觸發器來維持圍阻體氧氣濃度
	2.圍阻體控制氫氣濃度之裝置
19-018	1.反應器基座混擬土之材質
19-019	1.儀控設備在事故環境下之適存性
	2.偵測器在事故環境下之適存性
	3.提供核子事故下之環境及儀控設備能力之比較
19-020	1.事故發生機率之信心範圍
19-021	1.本章審查之依據
19-022	1.深度防禦觀念設計特性之應用
	2.熱移除系統能力之澄清
	3.不依賴交流電源補水系統(ACIWA)之壓力設定
19-023	1.核四廠與ABWR因應嚴重事故之不同點
19-024	1.停機時餘熱移除能力之檢討
	2.肇因事件之選取
	3.地震及颱風對核四廠之影響
19-025	1.列舉多樣性之反應水溫急停信號，以確保反應爐急停功能
	2.如何降低因備用硼酸系統動作後而導致之功率振盪
	3.廠外事件之預期暫態未急停(ATWS)事故發生之機率
19-026	1.發生SBO後，RCIC控制系統之能力
	2.發生SBO期間，濕度是否會造成電氣系統失效
	3.SBO事故序列分析圖之澄清
	4.SBO發生頻率及其導致爐心熔毀之機率
19-027	1.如何防止消防水誤動作而導致電氣設備之失效
	2.如何防止因火災導致短路的問題
	3.如何避免兩機組共用之設備因火災而同時失效
	4.火災肇因事件發生之頻率及其造成爐心熔毀之機率
19-028	1.核四廠因應ISLOCA之設計特性
	2.溫度對低壓管路耐壓強度之影響
	3.ISLOCA肇因事件之頻率及其造成爐心熔毀之機率
19-029	1.圍阻體中進出通路是否會在嚴重事故中產生熔融物堆積的現象
	2.停機時圍阻體未充氮的問題

編號	內　　　　容
	3.可燃氣體控制因人為失誤而失效的可能性
	4.COPS破裂閥設定及管路內充氮的問題
19-030	1.PSAR使用模擬程式之澄清
19-031	1.MAAP程式模擬氫氣燃燒現象之澄清
	2.壓力單位之統一
19-032	1.犧牲式乾井地板混凝土之選擇
	2.COPS因應壓力突增現象之能力
	3.補充說明使用分解事件樹(DET)之原因
	4.爐屑不均勻分佈在地板上之靈敏度分析
	5.CCI靈敏度分析(MAAP)之討論
	6.CCI分解事故之討論
	7.反應爐基座之耐壓強度
19-033	1.高壓熔融排放(HPME)分佈之假設
	2.因HPME造成壓力突升對COPS設定點之影響
19-034	1.澄清"移動式"穿越器
	2.COPS針對壓力突昇之考慮
	3.COPS開啟時間對外釋劑量之靈敏度分析
19-035	1.ACIWA系統使用消防水系統之澄清
	2.反應爐基座衝擊壓力之意義
	3.計算反應爐基座所使用材料性質之自然頻率
19-036	1.金屬與水反應所造成圍阻體之尖峰壓力
	2.儀器適存性極限之意義
	3.移動式穿越器之字義澄清
19-037	1.有關爐屑比熱及熱焓之選取

附錄 A　安全度評估

編號	內　　容
A-01	1.禁建區及 LPZ 限值 2.海嘯 3.電廠設計修改影響肇因事件之選取的因應對策 4.核電廠與人口中心距離
A-02	表 A1-6“；快速轉移至主發電機”之運轉員疏失為 0.5 的合理性
A-03	事故前及事故之人為疏忽差異
A-04	LCHP-PS-R-N 事件系統列雜質與 Corium 比例之決定
A-05	1.RPV 高壓氣爆之產生 2.表 AJ.3-1 中 Corium 之量如何得到
A-06	1.提供非線性分析之輸入值 2.非線性分析之電腦程式 3.表 AJ.16-1 之最大應力地點
A-07	最快的壓力增長率及相關事故。
A-08	1.水災分析是否適用於核四廠 2.燃料在事故時為水所掩蓋之水是否包括雙相的水 3.2/3 爐心高度為熔損之準則之理由
A-09	1.ADS 避免手動之適用性及風險評估 2.社會風險及個人風險之相關計算數據 3.核四廠設計和標準 ABWR 相異之比較
A-10	1.圖 A6-1 誤打成圖 A6-2 2.圖 A6-2 缺頁 3.在圖 AB.5-3 及圖 A11-3 中 IB 及 ID 明顯誤植
A-11	圖 AD.4-7 和圖 AD.4-8 錯誤
A-12	圖 AJ.15-2、AJ.16-2、AJ.16-3 等單位不統一且有誤
A-13	1.在 A6.2.3.2.1 之地震 PRA 的前五項事故序列和 A1、4.1.2 及 AC.10.2.1 不同 2.SBO 及 ATWS 之 CDF 在內容上有錯置，引述不當情形 3.圖 AC.6-1 中，地震事件樹之序列頻度為何均為零 4.圖 A1-11、A6-3、AC.10-3，為何 0.4～0.6g 群比 0.0～

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	0.4g 群有較低 CDF。
A-14	1.請確認反應爐內泵(RIP)之耐震值將高於 1.8g 2.表 AC.3-10 所列之組件是否全部屬於奇異公司之範圍，其中消防水泵之耐震值，奇異公司是否保証由 1.8g 提升至 2.8g 3.請將所有在地震故障樹切割組中出現之組件列入表 AC.3-11 4.電廠特性之 SSI 分析是否已完成？若完成請將結果反應在耐震度分析中，若尚未完成，請明訂完成時限
A-15	1.請提供 CDF 前 90%之重要事件序列的最小切割組 2.請提供表 A4-2 中，初判定"可忽略"之肇始事件的詳細計算及理由
A-16	1.請解釋消防 CDF 在表 AD.1-1、表 AD.4-2 和 A1-10 結果不同的理由 2.請提供第 AD.3-1 頁中 fire compartment ignition freguency 之詳細說明 3.請在圖 AD.-1 至 AD.4-6 中加入 CDF 數據
A-17	1.A13.1.1 節提到 external events 之 CET 詳述於 Attachments AC， AD， AE 及 AF，其中 Fire(AD)，Flood(AE)及 Typhoon(AF)均有專節討論 CET，唯獨 Seismic(AC)部份未提說明，宜補充之 2.A14.1 節第(3)點提到"The elevation of the release is conservatively assumed to be at the reactor building roof height."此一說法與台電公司於 86 年 11 月 14 日在原能會簡報之說法不符，宜澄清之。 3.A14.6 節說明 Attachment AJ 中所做的 Source Term 與 Containment Release Analysis 均以 US Standard ABWR 為參考廠，核四之重大改變如 SSE 由 0.3g 提升至 0.4g 等均未納入考量，此點應略述其影響並規劃修訂期限 4.AB.6.11.3.3.2 節提及 recovery of high pressure injection system 的機率列於 A.J.4.2 詳細說明，但該節並無任何定量結果以支持此節之數據 5.AG.8.4.1.8 節稱 Shutdown CET for subclass VIA similar to that for internal CET for class ID，但或然率之引用有

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	誤，例如在 class ID 中，RHR recovery prob. For core melt not arrested and no active injection to lower D/W 的情形是"0"，但 subclass VIA 引用時卻定為"0.8".(Ref. Pages AB-6-38 及 AG-8-13) 6.請說明 MAAP 3.0B-ABWR 程式是否經美國 NRC 審查認可？(AJ.1.1.2 節) 7.現階段分析之假設(如部份 containment 建築採 Limestone sand concrete，lower srywell flow 則採 Basaltic，RCIC room cooling 可維持 8 小時等)宜訂定確認時限。(AJ.1.2.1 節)
A-18	有關核四廠安全度評估所採用之肇始事件應和現有沸水式核電廠有所不同，其新增項目機率如何決定，請澄清。
A-19	根據歷史經驗，台灣北部沿海曾有海嘯現象，其風險評估是否涵蓋海水退降而導致冷卻水流失、其或危及爐心，請加以澄清。
A-20	簡報資料表 2 所列目標值是如何決定，和國內外現有類似電廠之比較結果，請補充說明資料。
A-21	目前使用國外的風險基準雖然較保守，但為了更能評估設計之安全性，建議仍另外列出本土數據以作為比較。

附錄B　整體可靠度分析

編號	內　　　　容
B-01	5.反應爐急停分析方法 6.故障模式分析方法 7.預防維護保護
B-02	反應爐急停，二次側部份之分析模式

附錄C　緊急計畫

編號	內　　　容
P-C-001	PSAR 的撰寫應明確說明在核四廠建廠過程中，台電公司欲(或必須)承諾的事項為主。本附錄 C 中所有的資料均屬一般性的描述，並未針對核四部分做出說明，宜大幅度的改寫。有關具體審查意見，將分別於相關章節中提出。
P-C-002	PSAR 內容若涉及其他單位(如：全委會作業執行室)時，台電公司應主動循正式程序協調定案後提供具體作法，而非逕行抄錄舊有資料提報。
P-C-003	PSAR 是一套必須公開陳列的電廠相關文件，且緊急計畫與電廠附近的民眾息息相關，所以在緊急計畫的編寫與審查上，絕對不能草率。但是從目前的附錄 C 內容來看，緊急計畫的規劃，並未獲得台電公司應有的重視。
P-C-004	報告中似乎提到或陳述太多原能會或全委會的工作與職掌，但台電公司本身應規劃或承諾的工作反而不甚清楚。例如: C-75～C-98 頁所提之近指中心似乎不應屬台電之職掌。台電公司應站在身為核反應器設施經營者立場，提出如何協助廠外組織進行輻射偵測與成立近指中心，以及可能做到的承諾。
P-C-005	報告中提到數項台電公司過去委託學術單位執行的計畫，如: C-65 頁提到 TEVACS，C-128 頁提到成立輻射傷害防治中心。台電公司是否表明會將這些已完成之成果落實在核四廠的緊急計畫中？但如何落實似乎沒有詳述。
P-C-006	報告中未提到如何建立因應電廠緊急事故時廠內及廠界之輻射偵測計畫及監測網。
P-C-007	隨著科技的進步，各項通訊、電腦設備早已日新月異，但本報告顯然仍抄襲舊有的資料，未作考慮更新。例如:無線電話(大哥大)、電視傳真、網路系統等均未列入(C-105 頁)。
P-C-008	台電公司應說明事故時如何整合機組狀況、劑量分佈、氣象條件、交通狀況、預估劑量等資訊，以提出廠外民

編號	內　　　　容
	眾劑量及防護行動建議。
P-C-009	台電公司應說明有關民眾宣導與教育的規劃及預定執行的方式。
P-C-010	報告中所用輻射單位宜全部改為新制 SI 單位。
P-C-011	報告中引用「準則」及其他資料時，未適當修正相關文字，致出現「本準則」、「詳如 5.1.2.8 要求」等字眼；C-93 頁之近指中心通報作業亦不符規定。整體而言，本報告之編寫及台電公司自行審查之作業似嫌草率。
P-C-012	本附錄 C 應補充製作目錄，以便於查閱。
P-C-013	本節應對電廠概況(如: 電廠系統、容量、廠區位置、地理環境、廠區附近人口分佈、道路狀況及氣象等)略加敘述。
P-C-014	本節所述法規依據不完整。
P-C-015	本節多為原則性說明，宜補充計畫架構摘要，並針對責任區分、組織分工等摘要說明。
P-C-016	本節有關緊急計畫區的訂定方法不正確。
P-C-017	有關緊急計畫區暫定為五公里之理由，請再加以詳細說明。
P-C-018	各重要應變單位多未指明負責人及其在原單位之職稱。
P-C-019	廠內各緊急工作隊、組、中心負責人及其代理人、編組成員與人數等，應予明確規劃、說明。
P-C-020	表 1-2 中未說明緊急民眾資訊中心之任務；執行 PASS 取樣之工作隊組亦應予確定。
P-C-021	未說明通知流程(包括: 負責通知之單位與人員、通知之時機與內容等資料)。
P-C-022	第二類緊急事故時，廠內緊急組織之動員時機與方式(局部或全體動員)、局部動員之工作隊組等應予明確訂定，表 2-1 並應對應修正。
P-C-023	本節請針對核四廠特性，補充說明各類事故之判定準則，如就核燃料、圍阻體等範圍，依儀表指示、系統狀況等訂定相關判定準則。
P-C-024	• 與醫院之協定中，附件一注重輻傷中心之建立，對緊急應變措施較無描述，附件二則只針對輻傷病患訂定

編號	內　　容
	急救方式，對一般傷患則付之闕如。 ・應補充提供與榮總簽訂之有效合約影本。
P-C-025	與醫院的協定只適用於台北榮總，對位居貢寮之核四廠恐嫌太遠，宜仿核三廠在廠區附近找一適當之醫療單位，與其訂定初步急救協定。
P-C-026	本節尚缺與軍方部隊之相關協定。
P-C-027	本節尚缺兩種以上通訊方式之說明、各單位負責連絡人員及定期通訊測試計畫等。
P-C-028	電廠須針對事故中緊急救援行動，如：傷患急救、設備搶修、事故評估及人員除污等依相關法規擬訂救援人員之緊急輻射曝露計畫，並指定或授權適當主管人員於事故中負責該等救援行動之核准與監督工作。
P-C-029	本節未說明疏散人員清點方式及交通工具、傷患緊急救護措施與設備等。
P-C-030	本節太簡略，請參照「核能電廠初期安全分析報告緊急應變計畫審查導則」(如附件)擬訂較具體之措施。
P-C-031	本節除應參照 III.4 節之審查意見補充說明外，並應補充分析所需軍、警、消防、交通等單位支援事項。
P-C-032	本節請補充說明初步傷患急救與除污之設備與能力。
P-C-033	本節請補充說明初步傷患急救與除污之設備與能力、核四廠附近醫院之名稱，以及傷患送醫之交通工具。
P-C-034	本節太籠統，請具體說明設備、人力及能量等資料。
P-C-035	本節請補充訓練課程內容、時數、頻次與考核方式等。
P-C-036	本節應配合Ⅲ.8 節，訂定廠外支援人員之訓練計畫。
P-C-037	本節請補充負責通知之人員、通知內容、時機及所需時間等。
P-C-038	C-63 頁，請以更直接的方式說明警報的內容、形式。
P-C-039	本節應實際就龍門電廠緊急計畫區內人口、訪客、道路、交通狀況、氣象等資料進行疏散分析。
P-C-040	本節請就疏散道路容量、緊急醫療能量等考量障礙內容、解決方式及期程等。
P-C-041	・無圖 15-1 ・應承諾將在 FSAR 提出時，明確標出可能使用的集結

編號	內　　　　　容
	點。
P-C-042	本節請大略說明構想中的評估設備、系統與方法，包括那些能力及必須滿足那些功能等。(參考 PSAR 附錄 A)
P-C-043	本節請大略說明構想中的評估設備、系統與方法，包括那些能力及必須滿足那些功能等。(參考 PSAR 附錄 A)
P-C-044	本節應具體承諾在 FSAR 中明確劃定輻射偵測路線及偵測點。
P-C-045	本節雖提及空中偵測但並無具體作業程序。
P-C-046	本節缺民眾訊息傳遞之說明。
P-C-047	請描述 PADES-1 的基本功能。
P-C-048	本節應依廠內各緊急控制場所之人員、設備、輻防等需求，明確劃定各場所設置地點及場所大小。
P-C-049	本節需要對緊急應變設施數據系統做更詳細的說明。
P-C-050	本節應依近指中心人員、設備及輻防需求，明確規劃設置地點及場所大小，並承諾會於 FSAR 明定近指中心地點。
P-C-051	在本節中，緊急應變計畫(編號 24)部分，電力公司承諾將於核四廠第一部機組初始燃料裝填前，編妥緊急計畫實施程序與第一次全廠緊急計畫演習方案，並執行全廠演習。但所稱編妥全廠緊急計畫演習方案一語，仍嫌過於籠統，宜提出具體之廠內、外緊急計畫演習之程序書，以供於第一次緊急計畫演習前，了解核四廠緊急計畫之完備性，及其對已往歷次核能電廠演習上獲取之經驗是否已有回饋。又附件三上所提出之 83 年版「核子事故緊急應變計畫」，包括目錄一至九內之說明，僅為實施要點，並不足以取代核四廠緊急計畫演習作業程序書。
P-C-052	本節請補充歷次演習日期、地點、參與人數、經費及評核結果摘要等資料，且針對評核所發現之缺失亦請提出具體改善措施。

附錄D　後端營運計畫

編號	內　　　容
D-01-1	拆廠作業立即拆除英文為何是 DECON？
D-01-2	大型商用反應器拆廠預估經費為三億美元(1997 年幣值)，須註明是否包含最終處置費。
D-01-3	龍門計畫預定拆廠工作流程中，燃料移出廠房之出處應該有所交待。
D-01-4	後端營運費用基金收支保管及運用辦法中基金之動支範圍，應包括後端營運工作之規劃研究、技術資訊評估分析、研究發展及溝通計畫，否則將來無法有效執行後端營運工作。
D-02-1	用過核燃料池貯存容量可容納未來運轉 40 年所退出之用過核燃料，但第 11 章 11-5-10 中之 radwaste building ventilation exhaust monitoring 中缺乏對 Kr-85、I-129 之放射性 range 之描述，請說明或補充。
D-02-2	設計容量 20，000 桶低放射性廢料之貯存倉庫為 40 年貯存期，在這期間是否也要執行低放射性廢料最終處置方案？
D-02-3	40 年貯存期，應該對貯存設施、盛裝容器之塗料規格有所規定，以確保安全貯存。
D-03-1	電廠拆除時各類廢料產量及處理情形，請參酌法規提出概要評估。
D-03-2	電廠拆除時之用過核燃料管理規劃(假設最終處置場未能如期完成時)，請提出說明。
D-04-1	依據 1990 年日本原子力學會誌(Vo1，32，No，5)所載「改良型沸水型原子爐(ABWR)之開發及實用化」一文，固體廢棄物產量每年 100 桶以下。
D-04-2	有關世界上已完成低放射性處置設施之國家，請增列日本青森縣六個所村處置設施。
D-05	低放射性廢料境內最終處置計畫之五個階段，請列出時程，並以甘氏圖表示其進度及關鍵途徑。
D-06	是否已將「可監測回收之貯存方式」(MRS)於後端營運中列入考慮，抑或將來需要時再另行審查。

附錄E　經驗回饋

編號	內　　　　　容
E-01	1.廠房設備管路等配置便利性
	2.海水管路腐蝕防護對策及檢查
E-02	國外核電廠除役經驗回饋
E-03	廠區建築物設計風速澄清

審查問題與答覆內容

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)：　N-17-001

問題章節(PSAR Section)：　17.1

初提日期(Question Date)：　1997.11.18

問題內容(PSAR Question)：

一、本節係有關設計和施工階段品保方案，PSAR 中列出台電，GE 及 S&W 公司之品保方案，請澄清在執行上適用之優先次序及台電公司將如何整合三者間之品保方案要求。

二、第 17.1B.1 節係GE之設計和施工階段品保方案；請澄清GE公司之下包商如Black & Veatch、Shimizu Hitachi、Toshiba、Foxboro 等之品保方案要求。

問題答覆(Responses)：

一、 1.台電、GE 及 S&W 公司之品保方案，在執行上分別適用於台電、GE 及 S&W 負責之工作範圍，三者並無優先次序及整合問題。

2.台電藉由採購合約之品保要求，確保承包商（如 GE、S&W）之品保方案能符合台電「核能工程品保方案」之要求（附圖一）。

二、 1.依據 NUREG-0800，PSAR 之品保方案須包括主要承包商之品保方案，但未要求將次包商之品保方案納入。龍門 PSAR 僅檢附台電與主要承包商 GE 及 S&W 之品保方案，已於 PSAR17.0 中說明。

2. GE Lungmen Team 之品保方案架構圖說明於 PSAR17.1B.1 之 Fig.1.2（如附圖二），GE 對其下包商之品保方案要求，對台電而言，屬承包商對次包商之要求，其管制方式如附圖一。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-17-002

PSAR Sections: Ch 17.3

Question Date: November 18, 1997

PSAR Question:

17.3 Reliability Assurance Program :

This section deals with RAP but it is combined in Appendix B. After reviewing Appendix B, it was found that the content and format are different from GESSAR or NRC requirements (such as SECY-89-013, 93-087, NUREG-1070 and NUREG-1503). Please clarify.

1. GE ABWR Design Phase RAP

   The D-RAP chapter includes the following :

   (1) Introduction
   (2) Scope
   (3) Purpose
   (4) Program Objectives
   (5) D-RAP Organization, responsibilities and function
   (6) Structures, systems and components' identification and prioritization
   (7) Design consideration : design review and reliability analysis during detailed design
   (8) Failure Mode definition (use NUREG/CR-5635 methodology)
   (9) Operation Reliability Assurance Procedures
   (10) Applicant or License Holder RAP
   (11) Implementation of D-RAP
   (12) Example Implementation

2. NUREG-1503 O-RAP

   O-RAP should include the following :

   (1) Monitoring of reliability functions
   (2) Reliability methodologies
   (3) Prioritization
   (4) Root cause analysis
   (5) Corrective action analysis
   (6) Corrective action implementation
   (7) Corrective action verification

(8) Nuclear power plant aging
(9) Designer feedback
(10) Program interfaces

PSAR Response:

Lungmen Preliminary Safety Analysis Report format and chapter title were established in Lungmen licensing activities kickoff meeting with Taiwan Power Company on November 20, 1996 which categorized Integrated Reliability Analysis (IRA) in Appendix B as equivalent to Chapter 17.3 of ABWR SSAR. NUREG-0800, Standard Review Plan, Chapter 17.4 describes Reliability Assurance Program.

It is recognized that the format of Lungmen PSAR Appendix B differs from the format given in the GE Nuclear Energy ABWR Standard Safety Analysis Report (SSAR), 23A6100. However, the same activities specified in the ABWR SSAR are included in Appendix B. Attached table provides a comparison of the activities specified in the ABWR SSAR with comparable activities in the PSAR Appendix B. Appendix B content and format is consistent with NRC requirements for Design Reliability Assurance Program (D-RAP).

The primary reasons for the format in Appendix B can be summarized as follows:

1. The PSAR Appendix B was formatted to more adequately reflect the Lungmen two-step licensing process (construction permit/operating license approvals) versus the ABWR one-step design certification process.

2. The PSAR Appendix B includes additional Lungmen specific requirements that were not included in the ABWR SSAR. Examples include Lungmen specific quantitative performance standards for plant safety and plant availability that must be demonstrated during the design and throughout the life of the plant.

3. As described in Appendix B Section B.1.1, Appendix B reflects the concept of an integrated reliability analysis (IRA) program which

ensures the safety and reliability are maintained as the detailed design evolves through the procurement and construction phase (D-RAP phase). In addition, pertinent information from the D-RAP phase is integrated into a single program for use by Taiwan Power Company during the operational life of the station (O-RAP phase).

4. The implementation process for the IRA program during plant operation (O-RAP) will be developed and included in the Lungmen FSAR. The process will include as a minimum the following activities:
   a) Reliability Performance Monitoring
   b) Reliability Methodology
   c) Problem Prioritization
   d) Root Cause Analysis
   e) Corrective Action Determination
   f) Corrective Action Implementation
   g) Corrective Action Verification
   h) Plant Aging
   i) Feedback to Designer
   j) Programmatic Interface

There is no change required to the PSAR from the above response.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

ROC AEC's PSAR Comment	ABWR SSAR	Where Covered in Appendix B PSAR
(1) Introduction	17.3.1 Introduction	B.1 Introduction
(2) Scope	17.3.2 Scope	B.1.2 Scope
(3) Purpose	17.3.3 Purpose	B.1.1 Purpose
(4) Objective	17.3.4 Objective	B.1.3 Program Objectives
(5) D-RAP Organization, Responsibility and Function	17.3.5 GE-NE Organization for D-RAP	B.1.4 Organization Interfaces
(6) Structure, System and Components Identification and Prioritization	17.3.6 SSC Identification/Prioritization	B.3.1.1.1 PRA & B.3.1.1.2 Unplanned Outage Analysis
(7) Design Consideration, Evaluation at the Detailed Design Stage by Design Reviews and Reliability Analysis	17.3.7 Design Considerations	B.3.4 Design Considerations
(8) Defining Failure Mode (Use NUREG/CR-5635 Methodology)	17.3.8 Defining Failure Modes	B.3.5 Defining Dominant Failure Modes of Risk-Significant SSCs
(9) Operational Reliability Assurance Activities	17.3.9 Operational Reliability Assurance Activities	B.3.6 Risk/Reliability Focused Maintenance Analysis
(10) Owner/License Holder Reliability Assurance Program	17.3.10 Owner/Operator's Reliability Assurance Program	B.5 Integration of the Reliability Analysis into the Program for Implementation during Operation
(11) D-RAP Implementation	17.3.11 D-RAP Implementation	B.3.7 Example Implementation
(12) Example of D-RAP Implementation	Example of D-RAP Implementation included in Section 17.3.11	Included in Section B.3.7

RESPONSES TO ROC-AEC's PSAR QUESTIONS

ROC-AEC Review Comments:

Please provide an implementation schedule of Integrated Reliability Assurance Program before CP.

Further clarification:

Lungmen IRA phase I evaluation is in progress. IRA phase I evaluation will identify the risk-significant SSCs and critical R items, unplanned outage analysis and qualitative system performance criteria allocation. The phase I evaluation is scheduled to complete at the end of April 1999. The phase I IRA evaluation will identify the Lungmen design can meet the IRA performance goals. The preliminary results of phase I evaluation will also provide the inputs for design evaluation and failure modes of risk-significant SSCs. The preliminary IRA evaluation will be updated in the design phase and these results will be described in the Final Safety Analysis Report (FSAR). The major tasks in the IRA program and targeted scheduled submittal date for TPC review is tabulated below. The schedule is based on the fuel loading date of Oct. 2003.

D-RAP Task Description	Where Covered in Appendix B PSAR	Target Submittal Date For TPC Review
(1) Introduction	B.1 Introduction	See PSAR
(2) Scope	B.1.2 Scope	See PSAR
(3) Purpose	B.1.1 Purpose	See PSAR
(4) Objective	B.1.3 Program Objectives	See PSAR
(5) D-RAP Organization, Responsibility and Function	B.1.4 Organization Interfaces	See PSAR
(6) Structure, System and Components Identification and Prioritization	B.3.1.1.1 PRA & B.3.1.1.2 Unplanned Outage Analysis	5/15/99
(7) Design Consideration, Evaluation at the Detailed Design Stage by Design Reviews and Reliability Analysis	B.3.4 Design Considerations	8/16/01

(8) Defining Failure Mode (Use NUREG/CR-5635 Methodology)	B.3.5 Defining Dominant Failure Modes of Risk-Significant SSCs	8/16/01
(9) Operational Reliability Assurance Activities	B.3.6 Risk/Reliability Focused Maintenance Analysis	8/16/01
(10) Owner/License Holder Reliability Assurance Program	B.5 Integration of the Reliability Analysis into the Program for Implementation during Operation	4/16/02
(11) D-RAP Implementation	B.3.7 Example Implementation	See PSAR
(12) Example of D-RAP Implementation	Included in Section B.3.7	See PSAR

There is no change required to the PSAR from the above clarification.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track　Number)：　　N-17-003 （再澄清版）

問題章節(PSAR Section)：　　17A

初提日期(Question Date)：　　1997.11.18

問題內容(PSAR Question)：

一、第一章組織第 1.2 權責區分第(7)項核發處「負責審查新建核能電廠有關…非破壞檢測之規畫、設計」，請說明將如何執行？

二、同節第 5 頁「品質處駐工地品保小組」負責(1)工地品質文件之審查(2)...(3)... 。請澄清駐工地品保小組將於何時成立，以執行其功能。

問題答覆(Responses)：

一、本公司核發處將依 NRC 核准之 1989 年版 ASME XI 法規規定審查。

二、本公司品質處駐工地品保小組未成立前之功能由該處負責，該小組預定 87 年第二季正式成立。

原能會審查意見(ROCAEC Review Comment)：

台電公司對「品質處駐工地品保小組」於何時成立之答覆為「預定 87 年第二季正式成立」。請說明何時成立？以及延後成立之原因。

台電澄清說明(Further Clarification)：

1. 駐工地品保小組原預定 87 年第二季成立，因適逢品質處轉換系統（由配電營運系統轉為綜合企劃系統），以及原主管副總經理於六月底退休，因此品保小組的派任案必須延到新主管副總經理指定後，才能提人選商議呈批，目前已在作業選派人員中。
2. 駐工地品保小組將於 87 年 10 月 15 日前成立。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： N-17-004

問題章節(PSAR Section)： Chapter 17

初提日期(Question Date)： 1997.11.18

問題內容(PSAR Question)：

一、 GE 2.0 p6 ， TPC 第二章
品保方案適用範圍中之 Reliability Class (Quality Class R)包含那些系統或組件，建議說明清楚以利依所訂規範執行。

二、 GE 2.0 p6 ， TPC 第二章
所稱之 Quality Class G 屬於一般商用級，絕大多數電廠 BOP 組件屬之。目前一般商用級產品符合 ISO-9001 系列者已是國際間之趨勢，反而此處之 Class G 組件未訂定任何規範依據，將如何採購？

問題答覆(Responses)：

一、 PSAR 第三章之 Table 3.2-1 "Classification Summary" 中，對 NI 、 BOP 之 Structures ,Components, and System 之 Quality Class 已有界定。台電並於採購規範中要求 NSSS,T/G 及 Phase II A/E 承包商必須建立 R-List 提送台電（詳如附件一,二），因此 Quality Class R 包含那些系統組件，將可由 PSAR 之 Table 3.2-1 及 R-List 界定清楚，同時 R-List 並不適合納入品保方案之適用範圍內說明。

二、 1. GE2.0.P6 中僅對 Class S,R,G 做定義，未提及 Class G 之 QA Requirements 。台電「核能工程品質保證方案」適用範圍不包括 Class G ，故在第二章中不敘述 Class G 之 QA Requirements 。

2. 對於 Class G 之 QA Requirements，台電於採購規範中已訂定，例如 NSSS Bid Spec.3.2.8.2C （附件三）。國內採購部份，並已正式公告必符合 ISO-9000 系列品質保證規定（附件四）。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： N-17-005（修正版並再澄清）

問題章節(PSAR Section)： 17A

初提日期(Question Date)： 1997.11.18

問題內容(PSAR Question)：

一、 第一章第(2)(11)節與第(4)(8)節內容相同，請澄清針對此項權責，施工處與品質處的分工情形。

二、 品保方案有關單位組織系統圖有核安會及核後端處，但在第 1.2 節權責區分中未見敘述，請補充。

三、 台電管理政策中聲明，要實施全面品質管理。
第 1.2.3 節中，品質處權責為推動品保方案之執行，並評估其成效，但在附錄一的品保方案權責區分表，品質處並未全面參與執行，尤其是與承包商品質之介面，僅第 1.3 節為協辦，其他章節都闕如，請補充。

四、 台電全面品質管理政策中聲明，核能安全有關項目及作業，除參照國家標準或國際標準，建立全品管體系，推行全面品質管理外，尚需依據原子能法規，實施核能品質保証方案，請說明全面品管之依據標準。

問題答覆(Responses)：

一、 對「新建核能電廠或新增核能機組工程之營繕工程授權範圍內之驗收」，施工處與品質處之間的介面關係為按“授權金額”做劃分。依據台電公司授權金額表（如附表），7500 萬元以下者由施工處驗收，7500 萬至 1 億 5000 萬元者由核火工處驗收，1 億 5000 萬元以上者由品質處驗收。

二、核能工程品保方案將配合增訂。

1.2.6 核安會：

審查核能電廠新建工程重大安全事項。

1.2.7 核後端處：

會同辦理核後端設施新建工程相關器材供應廠商之製程驗證，執行成效考核以及稽查作業。

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
八、材料、零件和組件之標識與管制	一、龍門施工處 龍門施工處器材、零件、組件及半成品標識及管制作業程序書	LMP-MTD-009	已發行	87.04.08

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
九、特殊製程管制	一、龍門施工處			
	1.龍門施工處電焊管制作業程序書	LMP-PPD-003	建立中	87.12.31
	2.龍門施工處熱處理管制作業程序書	LMP-PPD-007	建立中	87.12.31
	3.龍門施工處防護塗裝管制作業程序書	LMP-ARD-007	建立中	87.08.31
	4.龍門施工處化學清洗管制作業程序書	LMP-PPD-008	建立中	90.12.15
	5.龍門施工處非破壞性檢測管制作業程序書	---	待建立	88.06.30

三、 1. 台電全面品質管理政策聲明要實施全面品質管理之目的，在宣示品質工作必需要“全員參與、持續改善、顧客滿意”。因此，對於核能工程品保方案之執行，權責區分表 1.1.（2）中明訂核能工程相關單位均為主辦，必須按品保方案之權責劃分，辦理職掌內之工作。

2. 品質處在核能工程品保方案權責區分表所列的 89 項作業要求中，主辦者佔 30 項，協辦者佔 7 項，其餘未列主/協辦的作業項目，品質處係藉由評估、檢驗及品保稽查的功能來確保其符合品保方案的要求。

四、 1. 全面品質管理目前尚無國家標準或國際標準可供參照或依據。

2. 台電龍門施工處將依據台電全面品質管理政策聲明、10CFR50 Appendix B 及參照 ISO 相關國際標準，建立「龍門施工處全面品質管理手冊」，內容包括工程品質、工程管理、工安環保三部份，據以推行全面品質管理。

87.7.22 修訂問題答覆一之內容如下：

一、對「新建核能電廠或新增核能機組工程之營繕工程授權範圍內之驗收」，施工處與品質處之間的介面關係為按“授權金額“做劃分。依據台電公司授權金額表（如附表），7500 萬元以下者由施工處驗收，7500 萬至 1 億 5000 萬元者由核火工處驗收，1 億 5000 萬元以上者由品質處驗收。至於品質處必須獨立於驗收之外的品質保證任務，將依據品保方案十八章的稽查功能，另由擔任驗收工作以外之人員，對驗收工作執行品保稽查，以符合品保獨立自主的精神。

原能會審查意見(ROCAEC Review Comment)：

台電公司對本問題之第一小題：「...請澄清針對此項權責，施工處與品質處的分工情形」之答覆為：「...至於品質處必須獨立於驗收之外的品質保證任務，將依據品保方案第十八章的稽查功能，另由擔任驗收工作以外之人員，對驗收工作執行品保稽查...請台電公司澄清是否品質處之權責中，僅「驗收」乙項與工程實際執行之功能重疊？亦請檢視貴公司品質處之人員配置有多少是執行獨立自主之品保功能而非與其他部門平行作業？

台電澄清說明(Further Clarification)：

1. 由於龍門工程均採外包方式，故對整個龍門工程品保體系而言，品質處與施工處對該外包工程之「驗收」作業，均屬執行品保功能。如僅針對台電內部的品保體系而言，品質處則另負對台電內部作業品保稽查的任務。因此品質處人員不論是執行「驗收」或「稽查」工作，相對於整個龍門工程之品保體系，均屬執行品保功能，故無所謂品保功能與實際執行功能重疊之問題。

2. 品質處對驗收工作執行稽查時，將選派”負責該項驗收工作以外人員”擔任，以符合 10CFR50 App.B “稽查須由對受稽查部門不負直接責任之人員執行“之規定。

3. 基於前述說明，品質處人員不論是執行「驗收」或「稽查」工作，均係執行獨立自主之品保功能。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： N-17-006

問題章節(PSAR Section)： 17A

初提日期(Question Date)： 1997.11.18

問題內容(PSAR Question)：

一、第 2.2.5 節核能工程有關之供應商、承包商：
建立並實施品質保証方案；在 PSAR 17 章之 Attachment B 和 C，是奇異公司與石偉公司所建立之品保方案。
而 17.0 節中敘述"Plant constructor will work under TPC's QA program, and the QA program of the plant constructor is not addressed separately in this chapter"請於 PSAR 中補充闡明奇異公司及石偉公司之下包供應商、承包商適用品保方案將如何控制？

二、2.3.12 節之作業要求，有敘述核後端處之任務，但在第二章中職責區分並未列明，請補充。

問題答覆(Responses)：

一、1. 龍門工程將來的安裝/施工均由台電或台電之承包商負責，因此 PSAR 17.0 中敘述 Plant Constructor 將在台電的 QA Program 下運作，同時其 QA Program 不個別附入 PSAR 17 章中。
2. Plant Constructor 屬台電的施工承包商，與 GE 及 S&W 之 QA Program 並無直接關係，台電將藉採購合約之品保規範之訂定（例附件一），利用評估、審查、檢驗/稽查等方式，以確保 Plant Constructor 之 QA Program 及其執行，符合台電核能工程品保方案之要求（如附圖一）。
3. 台電對 GE 及 S&W 之下包供應商/承包商，將藉由與 GE 及 S&W 之合約之品保要求（如附件二）以品質巡查、檢驗、稽查等方式，確保其品保方案及其執行符合台電核能工程品保方案之要求。

二、核能工程品保方案將配合增訂：
2.2.6 核後端處
配合執行核能工程品保方案，以 確保核後端新建工程品質。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)：　N-17-007

問題章節(PSAR Section)：　Chapter 17

初提日期(Question Date)：　1997.12.17

問題內容(PSAR Question)：

17A 中對於建廠期間各項作業之品質管制由何部門負責？如何執行，應建立何種作業手冊(包括品管程序書及目錄)以及何時可提出此作業手冊，請澄清。

問題答覆(Responses)：

一、 PSAR 17A 台電公司「核能工程品質保証方案」中，對於建廠期間各項品質保証作業之負責單位 ，分別訂定於各章之「權責區 分」內，例如對採購器材之廠商製程驗証訂於 7.2 節，監督施工承包商執行檢驗計畫並執行檢驗或見証訂於 10.2 節，對不符合項目的管制訂於 15.2 節，對承包商及台電本本身各 項作業之稽查訂於 18.2 節。各相關單位依據「核能工程品質保証方案」各章的權責劃分及作業要求，另分別建立作業程序書據以執行各項品質保証作業。

二、 各單位應建立 之作業程 序書目錄及時程如附件。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： N-17-008

問題章節(PSAR Section)： Chapter 17

初提日期(Question Date)： 1997.12.17

問題內容(PSAR Question)：

17A 第 2.3.14 節，請澄清台電公司將來針對龍門工程如必須採用核能同級品零組件，管制作業的程序書何時可擬妥備用？其負責部門為何單位？請在 17A 中具體界定，以利執行。

問題答覆(Responses)：

一、 龍門工程如必須採用核能同級品零組件（DCGI）時，依據 85.8.16 龍門計畫處間分工及協調會議討論決議，台電的負責部門及核准程序如下：當合約商提出採用 DCGI 申請時，由履約單位（外購合約/核火工處；內購合約/龍門施工處）負責收件轉送核技處審查，其中品保部分由品質處會審，審查結果交由核火工處/龍門施工處 通知合約商。

二、 對核能同級品之技術審查作業，預定 87 年 4 月底前完成所需程序書。

三、 台電「核能工程品質保證方案」第七章將依據上述第一項討論決議，增訂相關單位對 DCGI 之權責及分工。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： N-17-009（修正版）

問題章節(PSAR Section)： Chapter 17

初提日期(Question Date)： 1997.12.17

問題內容(PSAR Question)：

17A 為一原則性宣示之符合 10CFR50 附錄 B 之承諾，具體落實還需要各相關部門建立完整之品保、品管和相關作業程序書，以供各部門執行時遵循之。請提供目前已建立、建立中和未來將建立之程序書的目錄及時程，以供本會進一步審查。

問題答覆(Responses)：

台電各相關單位執行「核能工程品質保証方案」，目前已建立、建立中和未來將建立之程序書目錄及時程如附件。
87.7.22 依據現況，更新問題答覆一之程序書目錄及時程如附件。

原能會審查意見(ROCAEC Review Comment)：

台電公司答覆所提供之「核能工程品質保證方案」建立時程中，品質處負責之「駐龍門工地品保小組作業程序書」之預計發行日期為該小組成立後六個月內。鑑於該小組截至 87.8.1 止尚未成立，請台電公司於核發建照前提出該小組作業程序書各項發行日期。

台電澄清說明(Further Clarification)：

駐工地品保小組各項作業程序書之發行日期詳如所附之編訂計畫表。

駐龍門工地品保小組作業程序書編訂計劃表

87.8.19

編號	程序書名稱	發行期限	備註
QD-F- 1.1	駐工地品保小組組織及其權責	87. 9.30	
QD-F- 1.2	駐工地品保小組工作月報	87.10.15	
QD-F- 4.1	廠商品質方案審查程序	87.10.15	
QD-F- 4.2	採購發包文件審查程序	87.10.15	
QD-F- 5.1	施工處作業程序書審查程序	87.10.15	
QD-F- 5.2	試運轉及啟動試驗程序書審查程序	88. 3.31	
QD-F- 6.1	駐工地品保小組文書管理程序	87.10.15	
QD-F-16.1	品質偏差趨勢分析程序	87.10.15	
QD-F-17.1	成套品質文件審查程序	88. 3.31	
QD-F-18.1	駐工地品保小組稽查程序	87.10.15	
QD-F-18.2	工地品質巡查程序	87.10.15	

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
一、組織	一、核火工處			
	1.核能電廠新建工程相關單位介面管理程序書	NFD-PELT-02(N)-T	已發行	85.03.21
	二、品質處			
	1.核能工程品質保證方案之發行、修訂及管制程序	QD-G-2.1-T	已發行	84.03.15
	2.行政院原子能委員會對核能工程違規事項/注意改進事項處理序	QD-G-18.2-T	已發行	83.05.06
	三、龍門施工處			
	1.龍門施工處全面品管委員會作業程序書	LMP-PMD-003	已發行	85.08.13

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
三、設計管制	17.核能電廠機械工程設計作業程序書	NED-M-3.1.30	已發行	82.12.02
	18.核能電廠機械工程設計審查作業程序書	NED-M-3.1.40	已發行	86.10.13
	19.核能電廠「核能蒸汽供給系統保證試驗程序書」及其試驗報(反應爐熱功率部份)審查作業程序書	NED-M-3.2.1	已發行	86.11.22
	20.核能電廠核能工程設計審查作業程序書	NED-N-3.1.40	已發行	86.11.10
	21.核能電廠模型之佈置設計審查作業程序書	NED-P-3.1.1-T	已發行	86.12.31
	22.核能電廠廠房佈置設計作業程序書	NED-P-3.1.30	已發行	84.09.15
	23.核能電廠廠房佈置工程設計審查作業程序書	NED-P-3.1.40	已發行	86.11.10
	24.核能同級品零組件檢證審查程序書	NED-E-3.1.2	已發行	87.04.22
	二、核火工處			
	1.核能電廠計畫型新建工程之設計與施工協調作業程序書	NFD-PELT-01(N)-T	已發行	85.02.15
	三、品質處			
	1.行政院原子能委員會對核能工程違規事項/注意改進事項處理序	QD-G-18.2-T	已發行	83.05.06

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
三、設計管制	四、龍門施工處			
	1.龍門施工處臨時設施機械工程設計管制程序書	LMP-TGD-004	已發行	85.09.02
	2.龍門施工處臨時設施土木工程設計管制程序書	LMP-CIV-012	已發行	83.12.31
	3.龍門施工處永久性工程設計文件施工可行性審查作業程序書	LMP-PMD-021	已發行	86.10.30
	4.龍門施工處工地變更管制作業程序書			
	五、燃料處	LMP-CIV-021	已發行	87.01.19
	1.燃料處核燃料製造廠設計變更報告審查作業程序書			
		PF-QC-02	已發行	85.02.26

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
二、品質保證方案	一、核技處			
	1.核能技術處專業技術人員考訓作業程序書	NED-A-2.1	已發行	86.07.08
	二、核火工處			
	1.核能工程檢驗人員訓練及資格銓定作業程序書	NFD-QLT-01(N)	已發行	84.11.17
	2.核能火力發電工程處暨所屬施工處年度自辦暨委托辦理訓練業程序書	NFD-PSN-01(G)	已發行	76.10.13
	三、品質處			
	1.核能工程品質保證方案之發行、修訂及管制程序	QD-G-2.1-T	已發行	84.03.15
	2.稽查評鑑人員考訓及資格銓定程序	QD-G-2.2	已發行	84.08.14
	3.核能工程檢驗人員考訓及資格銓定程序	QD-G-2.3-T	已發行	86.06.30
	4.作業程序書及簡式作業流程圖之發行、修訂及管制程序	QD-G-6.1	已發行	84.07.08
	四、龍門施工處			
	1.龍門施工處全面品管手冊制訂、頒行與修訂作業程序書	LMP-PMD-018	已發行	85.07.17

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
三、設計管制	一、核技處			
	1.核能電廠土木工程設計作業程序書	NED-C-3.1.30	已發行	84.08.14
	2.核能電廠土木工程設計審查作業程序書	NED-C-3.1.40	已發行	86.11.10
	3.核能電廠資訊工程設計審查作業程序書	NED-D-3.1.40	已發行	86.10.03
	4.核能電廠電氣單線圖設計審查作業程序書	NED-E-3.1.1	已發行	86.11.07
	5.電氣工程設計作業程序書	NED-E-3.1.30	已發行	82.10.23
	6.核能電廠電氣工程設計審查作業程序書	NED-E-3.1.40	已發行	86.11.04
	7.核能電廠開關場介面作業程序書	NED-E-3.3.1-T	已發行	83.10.04
	8.儀控工程設計作業程序書	NED-J-3.1.30	已發行	84.08.15
	9.核能電廠儀控工程設計審查作業程序書	NED-J-3.1.40	已發行	86.10.09
	10.龍門計畫工程整體設計及設計管理作業程序書	NED-L-3.1.2	已發行	86.04.30
	11.核能電廠業主設計準則編修作業程序書	NED-L-3.1.3	已發行	82.09.29
	12.工程設計文件審查管制作業程序書	NED-L-3.1.4-T	已發行	86.07.18
	13.新(擴)建核能發電工程計畫設計變更評估作業程 序書	NED-L-3.1.5-T	已發行	84.04.29
	14.核能電廠機械系統和設備設計審查作業程序書			
	15.核能電廠整體熱效率試驗報告審查作業程序書	NED-M-3.1.1	已發行	86.12.03
	16.新建核能電廠「一次圍阻體總洩漏率測試程序書」及其測試	NED-M-3.1.5	已發行	86.11.26
	告審查作業程序書	NED-M-3.1.13	已發行	86.11.21

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
四、採購文件管制	一、核火工處			
	1.外購(含內製)器材、廠家說明書及圖說稽催作業程序書	NFD-PECP-01(G)	已發行	80.09.24
	2.核能火力發電工程處國外採購或服務費用信用狀開狀與管理業程序書	NFD-PELP-01(G)	已發行	77.03.09
	3.核能火力發電工程處國外採購電匯付款之作業程序	NFD-PELP-02(G)	已發行	77.03.18
	二、品質處			
	1.採購文件審查程序	QD-G-4.1	已發行	82.05.27
	2.廠商品質方案查審查程序	QD-G-4.2	已發行	86.03.12
	三、龍門施工處			
	龍門施工處內購器材採購文件之編撰、審查、核准、發行、更及管制程序書	LMP-MTD-006	已發行	86.10.09

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
五、工作說明書、作業程序書及圖面	一、核技處			
	1.核能技術處圖面管理作業程序書	NED-A-5.1	已發行	84.12.21
	2.核能技術處預算管理作業程序書	NED-A-5.2	已發行	82.11.11
	3.核能技術處文件管理作業程序書	NED-A-5.3	已發行	86.07.01
	4.核能資訊營運系統開發/維護/管制標準作業程序書	NED-D-5.1	已發行	82.12.17
	5.核能技術處資訊課工作程序書	NED-D-5.3	已發行	86.10.27
	6.龍門計畫工程預算控制作業程序書	NED-L-5.1	已發行	84.01.18
	7.核能技術處工程帳單審核作業程序書	NED-L-5.2	已發行	82.08.24
	8.核能技術處技術服務委辦作業程序書	NED-L-5.3	已發行	86.07.10
	9.核能技術處支援電廠重大改善設計及專案工作程序書	NED-L-5.4	已發行	83.05.03
	10.龍門計畫工程駐顧問公司小組工作作業程序書			
	11.核能技術處採購及營繕工程文件傳送政風部門作業程序書	NED-L-5.5	已發行	82.12.17
	12.台電核能發電工程回饋系統作業程序書	NED-L-5.6	已發行	83.01.27
	13.新建核能電廠暫態與事故分析作業程序書			
	14.新建核能電廠暫態與事故分析管制作業程序書	NED-L-5.7-T	已發行	83.06.16
	15.新（擴）建核能電廠初期／終期安全分析報告編寫審查程序書	NED-N-5.1	已發行	83.09.13
		NED-N-5.2	已發行	83.09.13
		NED-N-5.3-T	已發行	84.08.24

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
五、工作說明書、作業程序書及圖面	二、核火工處			
	1.核能工程進度控制作業程序書	NFD-PELE-02(N)-T	已發行	86.06.10
	2.核能工程預算控制作業程序書	NFD-PELE-03(N)-T	已發行	86.06.22
	三、品質處			
	1.駐龍門工地品保小組作業程序書	---	待建立	小組成立後6個月內
	四、龍門施工處			
	1.龍門施工處作業程序書之發行修訂管制程序書	LMP-PMD-001	已發行	84.09.13

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
六、文件管制	一、核技處			
	1.核能技術處作業程序書發行管制程序書	NED-A-6.2	已發行	86.04.10
	2.核能技術處報告發行管制作業程序書	NED-A-6.3	已發行	86.04.21
	二、核火工處			
	1.核能火力發電工程處工程圖面、說明書及圖書雜誌管理作業序	NFD-PJM-06(G)	已發行	76.09.24
	2.核能火力發電工程處文書管理作業程序書	NFD-GNL-03(G)	已發行	84.07.08
	3.核能火力發電工程處英文文書處理與檔案管理作業程序	NFD-PJM-05(G)	已發行	79.02.10
	4.工程管理電腦化系統開發及維護作業程序書	NFD-INF-01(G)	已發行	84.07.27
	5.核能火力發電工程處作業程序書的發行及修訂管制程序	NFD-QLT-01(G)	已發行	83.12.27
	三、品質處			
	1.文書處理程序	QD-G-5.1	已發行	84.12.21
	2.作業程序書及簡式作業流程圖之發行、修訂及管制程序	QD-G-6.1	已發行	84.07.08

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
六、文件管制	四、龍門施工處			
	1.龍門施工處全面品管手冊制訂、頒行與修訂作業程序書	LMP-PMD-018	已發行	85.07.17
	2.龍門施工處英文文書管理作業程序書			
	3.龍門施工處作業程序書之發行修訂管制程序書	LMP-PMD-002	已發行	84.11.09
	4.龍門施工處圖面管理作業程序書	LMP-PMD-001	已發行	84.09.13
	5.龍門施工處承包商領借圖面資料管理作業程序書	LMP-PMD-012	已發行	85.07.25
	6.龍門施工處密件圖面管理作業程序書	LMP-PMD-004	已發行	85.07.25
	7.龍門施工處參考書籍、期刊、錄影(音)帶管理作業程序書	LMP-PMD-006	已發行	84.09.26
	8.龍門施工處發包工程施工說明書編撰、審查、核准作業程序書	LMP-PMD-007	已發行	84.03.08
	9.龍門施工處中文文書處理作業程序書			
	10.龍門施工處承包商文件審查作業程序書	LMP-ARD-006	已發行	86.09.11
		LMP-GAD-001	已發行	84.10.23
		LMP-PMD-023	建立中	87.12.31

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
七、採購材料、設備和服務之管制	一、核技處			
	1.核能同級品零組件檢證審查程序書	NED-E-3.1.2	已發行	87.04.22
	二、核火工處			
	1.核能工程計畫外購合約國內製交器材、設備製程查驗處理程書	NFD-QLTE-01(N)-T	已發行	85.03.27
	2.核能火力發電工程處一般設備管理作業程序書	NFD-GNL-08(G)	已發行	84.08.08
	三、品質處			
	1.廠商品質保證制度評鑑登錄作業程序	QD-G-7.1-T	已發行	85.02.09
	2.國外廠家製程驗證與管理程序	QD-G-10.1	已發行	87.05.03
	四、龍門施工處			
	1.龍門施工處內購器材規範管制及請購作業程序書	LMP-MTD-002	已發行	84.05.01
	2.龍門施工處內購器材之廠商製程驗證作業程序書	LMP-TGD-008	已發行	86.10.20
	3.龍門施工處內購器材驗收作業程序書	LMP-MTD-005	已發行	84.05.15
	4.龍門施工處內購器材採購作業程序書	LMP-MTD-004	已發行	84.10.27
	5.龍門施工處供應廠商評鑑和選擇作業程序書	LMP-NSS-004	已發行	87.06.03
	6.龍門施工處外購器材驗收作業程序書	LMP-MTD-013	已發行	87.03.17

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
十二、測試設備管制	一、龍門施工處 1.龍門施工處測試設備管制作業程序書	 LMP-PMD-020	 已發行	 86.03.25

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
十三、搬移、儲存和運輸	一、核火工處			
	1.核能火力發電工程處內外購器材倉儲作業程序書	NFD-GNL-04(G)	已發行	84.08.08
	二、品質處			
	1.新核燃料運輸查核要點	QD-G-13.1	已發行	82.04.24
	三、龍門施工處			
	1.龍門施工處器材裝卸與搬運管制作業程序書	LMP-MTD-008	已發行	87.04.08
	2.龍門施工處器材儲存及管制作業程序書	LMP-MTD-007	已發行	87.04.09
	3.龍門施工處外購器材包裝和運輸管制作業程序書	LMP-MTD-012	已發行	87.06.27
	4.龍門施工處器材清潔和保存管制作業程序書	LMP-MTD-010	建立中	88.06.30

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
十、檢驗	一、核技處			
	1.工程設計及技術服務工作驗收作業程序書	NED-L-10.1	已發行	85.06.27
	二、核火工處			
	1.營繕工程驗收處理程序書	NFD-QLT-02(G)	已發行	84.02.23
	三、品質處			
	1.營繕工程驗收處理程序	QD-G-10.2	已發行	82.06.16
	2.國外廠商製程驗證與管理程序	QD-G-10.1	已發行	87.05.03
	3.核能發電工程權責核能檢查管制程序	QD-G-10.3	待建立	87.12.31
	四、龍門施工處			
	1.龍門施工處檢驗辦法作業程序書	LMP-QLD-003	建立中	87.07.31

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
十一、試驗管制	一、龍門施工處			
	1.龍門施工處安裝後試驗計畫作業程序書	LMP-ICD-003	已發行	86.12.07
	二、核發處			
	1.運轉前試驗及起動試驗管理程序書	---	待建立	聯試小組成立後9個月內
	三、核安會			
	1.核安會審查作業程序書	NSC-2.0	已發行	82.07.10

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
十六、改正行動	一、品質處			
	1.改正行動處理程序	QD-G-16.1-T	已發行	84.07.26
	2.品質通報處理程序	QD-G-16.2	已發行	82.11.10
	3.行政院原子能委員會對核能工程違規事項/注意改進事項處理程序	QD-G-18.2-T	已發行	83.05.06

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
十七、品質保證紀錄	一、核技處			
	1.核能技術處品質紀錄及文件管制作業程序書	NED-A-17.1	已發行	86.06.12
	二、核火工處			
	1.核能品質保證紀錄管制作業程序書	NFD-QLT-02(N)	已發行	87.07.10
	三、品質處			
	1.品質紀錄管理作業程序	QD-G-17.1	已發行	85.05.20
	2.文書管理程序	QD-G-5.1	已發行	84.12.21
	3.品保紀錄目錄及移交管制程序	---	待建立	87.12.31
	四、龍門施工處			
	1.龍門施工處品保紀錄管制作業程序書	LMP-PMD-022	建立中	87.08.31
	五、燃料處			
	1.燃料處核燃課品質紀錄管理作業程序書	PF-QC-03	已發行	87.02.06

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
十八、稽查	一、品質處 1.核能品質保證稽查程序	QD-G-18.1-T	已發行	84.07.26

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)：　17-010

問題章節(PSAR Section)：　Chapter 17

初提日期(Question Date)：　1997.12.23

問題內容(PSAR Question)：

第 17 章品保方案內所曾引用法規及標準之版次，請明確界定，以利執行時遵循之。

問題答覆(Responses)：

1. 龍門計畫適用之法規/標準及 其版次，於 PSAR 第一章之 1.8.2 "Applicability of Codes and Standards"中已有界定。
2. 台電「核能工程品質保證方案 」中曾引用之品保法規/標準，用於龍門計畫之適用版次，將以附註或附錄方式於品保方案中補充說明。
3. GE 公司品保方案內所引用法規及標準之版次，已說明於 Subsection 17.1B.2 、 Table 17.1B-1 及 Attachment 17B 之 Section 24.0 。 S&W 公司品保方案內所引用法規及標準之版次 ，已說明於 Subsection 17.1C.2 及 Attachment 17C 之 Appendix Ⅶ。

台電再澄清說明：

1. ISO-9000之版本係採用NOA前已正式發行之最新版本。對可靠性項目，其QA Program須符合ISO-9001 1994年版，其Software QA Program 須符合 ISO-9000-3 1991年版。
2. 台電「核能工程品質保證方案」，將於增訂之附錄四中明訂ISO-9000之版本。
3. GE公司品保方案，已於 PSAR的 Attachment17B之Section 24.0中明訂ISO-9000之版本
4. S&W 公司品保方案，將於品保方案之相關章節註明所引用 ISO-9000 之版本。

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
十四、檢驗、試驗和運轉狀況	一、龍門施工處			
	1.龍門施工處設備安裝檢驗及試驗狀況標示作業程序書	LMP-ELD-006	建立中	87.06.30
	二、核發處			
	1.運轉前試驗及起動試驗管理程序書	---	待建立	聯試小組成立後9個月內

核能工程品質保證方案相關作業程序書目錄

品保方案章別	程序書名稱	程序書編號	已發行/建立中/待建立	發行日期/預計發行日期
十五、不符合材料、零件或組件	一、品質處			
	1.廠家品質不符報告審查程序	QD-G-15.1-T	已發行	86.04.21
	2.缺陷通報管制程序	QD-G-15.2-T	建立中	87.09.30
	二、龍門施工處			
	1.龍門施工處不符合報告(NCR)處理作業程序書	LMP-QLD-001	建立中	87.07.31
	三、核安會			
	1.核安會審查作業程序書	NSC-2.0	已發行	82.07.10

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)：　17-011

問題章節(PSAR Section)：　17A, 17B, 17C

初提日期(Question Date)：　1998.01.19

問題內容(PSAR Question)：

1. Please have GE provide more detailed information of Lungmen organization chart (offsite vs. on site chart). (review criteria : SRP 17, II.1A5)

2. Please provide additional explanation on TPC QA manager power to stop work. (review criteria : SRP 17, II.1B4)

3. The policy statement in the QA Plan should be provided by the president or vice president of a company. In GE case, it was done by S. A. Hucik, General Manager of Lungmen Project and the signature was missing. Please clarify. (review criteria : SRP 17, II.1C1)

問題答覆(Responses)：

1. Attachment 1 is GE Lungmen Project Organization chart. The Taiwan Office on the chart is a part of the GE Lungmen Project Organization engaged in activities that are off-site to the Lungmen NPS site. Presently, GE is not participating in any activity that is on-site at the Lungmen NPS site. Document 31113-OA18-0001 included in Attachment 17B will be revised to include the GE Lungmen Project Organization. The chart will be updated in the future when GE will participate in the on-site activities.

2. 有關停工之職權，台電將於「核能工程品保方案」第一章 1.3.6 (5) 節品質處駐工地品保小組之權責內增列：
⑥對工地發現的重大品質缺失，向施工處提出停工要求，以免缺失擴大。

3. The GENE company-wide policy statement, which is provided in document NEDO-11209 of Attachment 17B carries the signature of Vice President and General Manager. The document NEDO-11209 also includes a letter from the USNRC accepting it as acceptable QA topical report. The document 31113-OA18-0001 included in Attachment 17B carries the signature of General Manager, S. A. Hucik, because the document is specific in scope of application to the Lungmen Project, i.e., as stated in Statement of Policy and

Authority, the document describes the quality assurance program which is to be used for the control of the design, procurement, and manufacture of equipment and components for the Taiwan Power Company (TPC) Fourth Nuclear Plant, Lungmen Units 1 and 2, Nuclear Island (NI) as specified in the contract between Taiwan Power Company and the General Electric Company (GE).

The signature of S.A. Hucik on Page 1 of the document 31113-OA18-0001 and the signature of Philip Novak, Manager, NPP/Lungmen Quality, on the cover sheet were not included on the electronic version of the document prepared by WORD software. The PSAR will be revised to include these pages with signatures as Attachment 2.

The PSAR will be revised as indicated in the responses above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： 17-012

問題章節(PSAR Section)： 17 A附件第六章 –

初提日期(Question Date)： 1998.5.15

問題內容(PSAR Question)：

一、台電品保方案第六章第 6.3.1(1)節文件管制範圍，請依據 SRP 17.1.II.6A1, Page 17.1-14 之要求，補充涵蓋 "As-built documents, Topical reports 及 Nonconformance reports. Design documents (e.g. analyses) including documents related to computer codes. "

二、台電品保方案第六章第 6.3.1(4)節，各類文件應視需要建立一總覽表。請依據 SRP 17.1.II.6B2 要求補充 "When such a list is used, it should be updated and distributed to predetermined responsible personnel "。

問題答覆；

參照審查意見，修訂品保方案相關章節如下，其中 Topical reports ，台電無該項文件，故不列入：

6.3.1(1) -------，文件種類至少應包括：

品保方案。

設計文件(諸如：計算書、圖面、規範、分析書)包含與 computer codes 相關之文件。

採購文件。

作業程序書、說明書。

竣工文件(As-built documents)。

安全分析報告書。

不符合報告。

6.3.1(4) -------

各類文件應視需要建立一總覽表, 以標明最新現況。 總覽表必須維持更新，並分送給需要使用之人員。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： 17-013

問題章節(PSAR Section)： 17 A 附件第七章

初提日期(Question Date)： 1998.5.15

問題內容(PSAR Question)：

一、台電第七章採購材料、設備和服務之管制，應包含設計、採購、 QA 組織間介面關係之權責歸屬（根據 SRP 17.1.II.7A1 ），請澄清。

二、台電第七章第 7.2.3 節品質處權責為國內供應廠商品質能力評鑑及國外供應商製程中檢驗、評估管制品質成效。根據 SRP 17.1.II.7A2 ， QA 單位應根據 Purchase Order 要求，對供應商作業(activities)作相關驗證(Verification)。請補充。

三、根據 SRP 17.1.II.7B6 ，品保方案在「採購材料、設備和服務之管制」的章節中，應承諾滿足 Regulatory Guide 1.38 的要求，請補充。

問題答覆(Responses)：

1. 台電的採購管制作業，其設計單位(核技處)、採購單位(核火工處、龍門施工處)、及品質單位(品質處)之權責區分，分別規定於品保方案之 7.2.1~7.2.4 節。
2. 依據 SRP 17.1 II.7A2，品保單位係參與 Verification，而 Verification 方式包括 Audits 、 Surveillance 、 or Inspection。品保方案 7.2.3(2)規定品質處負責國外供應廠商製程中驗證，及 18.3.1 規定品質處須對國外供應廠商執行稽查，均屬 Verification 作業。
3. 品保方案增訂"7.3.7 採購管制作業須符合 Regulatory Guide 1.38 有關對採購器材之包裝、運輸、接收、倉儲和搬移的要求。"

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： 17-014

問題章節(PSAR Section)： 17 A，17 B第二章

初提日期(Question Date)： 1998.5.15

問題內容(PSAR Question)：

根據 SRP 17.1.II, 2A(1e)，品保方案需承諾考慮特殊技術、環境條件、技術或製程。在 17A, 17C 台電及奇異公司的品保方案第二章中未提及環境條件的考量，請補充。

問題答覆(Responses)：

台電、GE及S&W公司品保方案，對須考慮環境條件的承諾，分別敘述於PSAR Attachment 17A之2.3.3(2)，17B之Reference 1之2.1，及17C之Section 2之1.2.2(詳如附件)。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)：　17-015

問題章節(PSAR Section)：　17 A

初提日期(Question Date)：　1998.5.15

問題內容(PSAR Question)：

SPR 17.1.II.8-13 提及品保十八條中第八條到第十三條準則，對所有工程專案各階段之品保活動予以規範。但台電品保方案第八到十三章的內容，侷限於施工處的權責，即在施工現場的品保管制。請台電公司以業主的立場，對於駐廠施工以外所有原供應廠商相關部份予以補充，以符合本法規要求。

問題答覆(Responses)：

台電品保方案不宜將供應廠商的品保作業納入予以規範。供應廠商須按台電採購規範的品保要求建立其品保方案並據以執行。台電依據品保方案第二章、第四章、第七章、第十五章、第十八章之規定，訂定採購品保要求，並藉由對供應廠商品保方案的審查、製程驗証、不符合項目的管制及稽查等，以確保其品質符合台電品保方案之要求。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： 17-016

問題章節(PSAR Section)： 17 A 第十、十一章

初提日期(Question Date)： 1998.5.15

問題內容(PSAR Question)：

一、台電公司第十章 10.3.1(4)「必要的停留檢驗點需於適當文件內註明」。SRP 17.1.II.10C2 要求「檢驗點非經指定之檢驗員驗證，不得進行」，請台電公司品保方案第十章加以補充，以符合要求。

二、SRP 17.1.II.11B1d 要求「必要的停留檢驗點要經由業主、承包商或檢驗員作檢証」，請台電公司品保方案第十一章加以補充，以符合要求。

問題答覆(Responses)：

參照審查意見，修訂品保方案相關章節如下：

10.3.1(4) 必要的停留檢驗點須於適當文件內說明。停留檢驗點非經指定的檢驗員檢驗，不得進行後續的工作。

11.3.1(2) 試驗程序須包含試驗前準備事項、所須試驗儀器、合適的試驗環境、須由業主、承包商或檢驗員驗証之必要停留檢驗點，以及依據設計文件所訂的試驗要求及合格標準。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： 17-017

問題章節(PSAR Section)： 17 A 第十四章

初提日期(Question Date)： 1998.5.27

問題內容(PSAR Question)：

SRP 17.1.Ⅱ.14.3 要求對安全有關之相關檢驗、試驗和運轉狀況，如有更動應建立程序書予以管制，且需有如原審查及批准之同等級管制。請台電澄清。

問題答覆：(Responses)

檢驗、試驗和運轉狀況等影響品質之作業，依據台電品保方案 5.3.1 節規定，負責單位將會建立程序書或說明書予以管制。依據台電品保方案 6.3.1 節規定，該等程序書或說明書均屬管制文件，如需更動時，必需循原審核程序辦理修訂後，才能據以執行作業。

編號(Track Number)： 17-018

問題章節(PSAR Section)： 17 A第十五章

初提日期(Question Date)： 1998.5.27

問題內容(PSAR Question)：

1. 按 SRP 17.1.Ⅱ.15.5之要求，不符合報告要經由品保組織週期性的分析品質趨勢對重大的結論需提報高層主管作評估及審查。請台電澄清。
2. 台電品保方案第十五章第 15.2.1(4)、15.2.2、15.2.3(1)節敘述有關器材之追蹤管制，分別由施工處、核火工處及品質處三單位負責。請依據組織圖說明品質處在本項職責之定位。

問題答覆：(Responses)

1.龍門計畫之不符合報告，主要係由供應廠商於製造過程，或施工承包商於工地作業中產生，因此該供應廠商及施工承包商之品保單位須對其產生之不符合報告分析品質趨勢，及對重大的結論提報其高層主管作審查、評估。台電則審查、追蹤管制廠商及承包商通報之不符合報告之處理情形。

2.以整個龍門計畫而言，施工處、核火工處及品質處三單位對廠商通報之不符合報告的追蹤管制作業，均屬執行品保功能。以台電核能工程單位組織系統圖而言，品質處則是站在品保單位的立場，藉由稽查方式，確保施工處及核火工處對不符合項目的追蹤管制作業，能符合台電品保方案第十五章的要求。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track　Number)：　　17-019

問題章節(PSAR Section)：　　17 A 第十六章

初提日期(Question Date)：　　1998.5.27

問題內容(PSAR Question)：

SRP 17.1.Ⅱ.16.2 要求品保單位書面認同(documented concurrence)改正行動的適切性(adequacy)。請台電澄清。

問題答覆：(Responses)

品保單位書面認同改正行動的適切性之規定，已訂定於台電品質處作業程序書 QD-G-16.1-T「改正行動處理程序」中（詳如附件）。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： 17-020

問題章節(PSAR Section)： 17 A 第十七章

初提日期(Question Date)： 1998.5.27

問題內容(PSAR Question)：

SRP 17.1.Ⅱ.17.4 要求品保記錄儲存設施必須符合 R.G. 1.88 的規定，包括對火災防範的考量。請台電澄清。

問題答覆：(Responses)

台電品保方案將配合增訂：

17.3.1(10) 核能安全有關品保 錄之儲存設施，必須符合 R.G.1.88 的規定，包括對火災防範的考量。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： 17-021

問題章節(PSAR Section)： 17 A 第十八章

初提日期(Question Date)： 1998.5.27

問題內容(PSAR Question)：

1. SRP 17.1.Ⅱ.18A4 列出稽查經常被忽略的部份，其中(a)與(b)二項與初期安全分析報告階段有關，為廠址特性及早期採購有關之管制。請台電澄清。
2. SRP 17.1.Ⅱ.18b1 要求稽查數據要由品保單位加以分析，結論報告應提出相關品質問題及品保方案之有效性，包含是否需對有問題的部份重作稽查，並應送管理階層審核及評估。請台電澄清。

問題答覆：(Responses)

1. 台電品保方案 18.3.1(1)配合修訂如下：

   18.3.1(1)稽查計畫：

   配合工作的進展，訂定稽查計畫。稽查範圍須包括 NUREG-0800 之 17.1.Ⅱ.18A4 所列出稽查經常被忽略的部份。
2. 依據品質處作業程序書 QD-G-18.1-T「核能品質保證稽查程序」之 6.6.1(三)、(四)、(五)， 6.7 及 6.8.1(詳如附件)。品質處稽查報告內容包含趨勢分析、重大品質偏差事項及綜合建議等，稽查報告並送管理階層審查及核閱，符合 SRP 17.1.Ⅱ.18b1 要求

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 18-001

PSAR Sections: Chp 18

Question Date: December 24, 1997

PSAR Question:

1. Please clarify that the compatibility of the arrangement for controls and the displays of associated parameters have been taken into M-MIS design's consideration. Refer to Table 18.4-1 and Table 18.4-3 as an example, the locations for fixed-position controls and the locations for monitoring the associated fixed-position display parameters shall be arranged so as to facilitate operator's work.

2. Please describe the methodology of information presentation on display units. Whether the arrangement of information presentation will be based on priority, functionality and the sequence of operation or not.

3. Please clarify that a help system for recovering from a human error operation will be included in the HSI design.

PSAR Response:

1. Compatible arrangement of controls and displays have been considered in the M-MIS design. Part II(2)(e)(i) of PSAR Table 18.7-1 on page 18.7-9 states that lessons learned from previous nuclear plant HSI designs as defined by the attachment to Table 18.7-1 shall be adequately addressed for Lungmen. Part II(3) of the attachment to PSAR Table 18.7-1 on page 18.7-24 states that controls and related displays should be in close proximity, readily associated, conveniently used with one another, placed in an obvious and consistent order, and arranged in functional groups.

2. The methodology for information presentation on display units has not been determined at this stage of the M-MIS design. The methodology will be determined on the basis of plant operating experience reviews and ongoing analyses of functional requirements, operator tasks, and

human reliability, and it will be described in the FSAR.

3. The HSI design will include features that help in recovering from maloperation of the plant as well as recovering from maloperation of the HSI equipment (e.g., the touch-screen user interface). The root cause of maloperation might be human error, but the HSI features will not be capable of making that diagnosis. An example of help for potential human error operation of the plant is the capability to monitor plant Technical Specification for violations of Limiting Conditions of Operation (LCO) and present recovery actions to the operator. A help system for potential human error operation of the HSI equipment has not been fully specified at this stage of the design although it will be based, in part, on the HSI design guidance of NUREG-0700, Rev. 1, Volume 1, Part 2, Section 2.7 concerning prevention, detection, and recovery (correction) of errors.

No changes will be made to the PSAR as a result of the responses to the above questions.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **18-002**

PSAR Sections: **Ch 18**

Question Date: **December 26, 1997**

PSAR Question:

1. Please explain how the human factor considerations will be factored into the Lungmen design and construction for TPC personnel especially the operations people.

2. The human factor consideration is concentrated mainly in the main control room and control panel areas. However, things like NRC position on Local Valve Position Indication as outlined in Section 18.6 of NUREG-1503 was not addressed. Please explain how it will be implemented in the future.

3. The original ABWR design requires only one operator to operate the main control panels In order to satisfy the current regulations, GE modified the requirements to have two operators. The work interface between these two operators should therefore be clearly defined and human factor consideration should also be given to the work load analysis or verification.

4. GESSAR Section 18.4.2.2 mentioned that there will be a laydown space of reference documents or procedures for the operators in the main control panel but no such description was provided in the PSAR. Please clarify.

5. This chapter does not have the Automation Design as mentioned in GESSAR Section 18.4.2.6. Please explain the reason.

PSAR Response:

1. As mentioned in PSAR Section 18.1, human factors principles are being incorporated in the Lungmen design consistent with the HFE program model of NUREG-0711. Plant personnel addressed by the HFE

program include licensed control room operators, non-licensed operators, and shift supervisors. Other personnel performing tasks directly related to plant safety are also considered. These personnel can include instrument and control (I&C) technicians, maintenance personnel, radiological protection technicians, and engineering support personnel. In particular, human factors considerations (e.g., anthropometrics, crew organization, and lessons learned from past operating experience) concerning operators, shift supervisors, and maintenance personnel, are being factored into the Main Control Room design by having members of Taiwan Power Company with these qualifications and expertise participate on the Lungmen control room design engineering team. Human factors considerations for various personnel are embodied directly in human factors engineering (HFE) requirements published in the U.S. by EPRI for Advanced Light Water Reactors (ALWRs). Lungmen NPS is being designed in accordance with these ALWR HFE requirements.

No change to the PSAR will be made as a result of the response to this question.

2. Local valve position indication as it relates to Chapter 18 on human factors engineering is addressed in Part V(1)(g) of Table 18.7-1 on page 18.7-16. Part V(1)(g) states "Analysis of functions and tasks shall support the basis for valve position indication at local control stations, and remote valve position indication for manually operated valves."

   No change to the PSAR will be made as a result of the response to this question.

3. The HFE process for Lungmen addresses U.S. regulatory issues concerning the number of operators at the main control console (Sections 18.2.2.2 and 18.2.3 of NUREG-1503). The work interface between the two operators is being defined, and human factor considerations are being given to operator workload analysis and verification. The work interface between the operators is influenced by nearly every element of the HFE process including staffing, function allocation, task analysis, human-system interface (HSI) design, procedures, and verification and validation. Operator staffing levels

(including roles and responsibilities) are considered during initial function allocation as part of the process of establishing an appropriate level of automation. The allocation of functions to operators considers human strengths and limitations, and the operator skills needed. Analysis of the allocations confirm that operators can perform their allocated functions and tasks while maintaining situation awareness, acceptable workload, and personnel vigilance [PSAR Table 18.7-1, Parts IV(1)(c and e), pg. 18.7-14]. Task Analysis addresses job design considerations such as workload, staffing, and communication requirements (i.e., the number of operators, their technical specialties and specific skills, the form and content of communications, and other personnel interaction required when more than one person is involved) [PSAR Table 18.7-1, Parts V(2)(c)(v and viii), pg. 18.7-17]. The HSI design implements operator interface requirements derived from Task Analysis [PSAR Table 18.7-1, Part VI(1)(b), pg. 18.7-19]. Procedures reflecting the results of Task Analysis are used in the Human Factors Verification and Validation process which confirms that the HSI can be operated using the established MCR staffing levels [PSAR Table 18.7-1, Part VII(1)(d)(ii), pg. 18.7-21]. Human factors performance measures for evaluating operator task performance test results include work interface factors such as crew workload, communications, and coordination [PSAR Table 18.7-1, Parts VII(1)(f)(iv and v), page 18.7-22].

No change to the PSAR will be made as a result of the response to this question.

4. The Lungmen NPS will be equipped with a laydown space so that hard copies of procedures and other documents, required by the operators during the performance of their duties, can be viewed.

   The PSAR, section 18.4.2.2, end of the 3rd paragraph, will be modified to include the above sentence.

5. The automation function originally described in GESSAR Subsection 18.4.2.6 is contained in PSAR Subsection 7.7.1.5.2. The description of this function was moved from Chapter 18 to Chapter 7 since it is more accurately a plant process computer system function than a human

factors engineering issue.

No change to the PSAR will be made as a result of the response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 18-003

PSAR Sections: Ch. 18

Question Date: March 20, 1998

PSAR Question:

1. Please provide comparison of differences in main control room operation between Lungmen NPP and conventional NPP.
2. During the detailed design of Lungmen in the future, please also include the User Manual which should consist of instructions on the maintenance of interface systems and how to correct software system errors to assure the correctness of operation.
3. Please clarify whether the I&C design of Lungmen includes self-diagnostics and display function of software errors. Also, if Lungmen employs symptom oriented incidents control design, then during the design of the information display of the error diagnostics, considerations should be given to how to assist the diagnostics work.
4. In the HFE design of Lungmen, considerations should be given to the specific features of the user population such as anthropometric data, vision field, human capability and limitation in vision and control, etc. to maximize the operator performance.
5. Please explain how the operator response capability and incidents handling capability can be assured at Lungmen when more automation is employed.

Response:

1. Section 18.4.2.1 of PSAR Chapter 18 lists design features of the Lungmen main control room. Some of these features directly concern operations and these particular features are not found in conventional nuclear power plant control rooms. There are additional operational features of the Lungmen main control room not found in conventional nuclear power plant control rooms. The following provides a composite list of several operational features of the Lungmen main control room that conventional nuclear power plant control rooms do not have.
   a) Automation of plant startup and shutdown evolutions such that a single operator at one control console can maneuver the unit from application of turbine gland sealing steam to rated power operation.

b) Touch-screen displays for safety system control and monitoring (including alarms) and touch-screen displays for non-safety system control and monitoring (including alarms).

c) Normal, abnormal, and emergency operating procedures displayed electronically, in diagrammatic form (e.g., flowcharts) with imbedded dynamic parameter indications and alarm information. [*Note: This feature is an operator aid, for use at operator discretion. Unit operation is not dependent on its use.*]

d) Operator display designs integrated with switching and tagging operations so that components "tagged" out of service are displayed in a distinctive manner.

e) Annunciator and alarm design based on a "quiet, dark board" concept (i.e., no audible or visual alerts should be active when the plant is operating normally at full power, with all systems in their normal configuration.). Alarm prioritization, filtering, and suppression enhance meaningful alarm presentation and reduce the amount of information that operators must be concerned with during abnormal conditions.

f) Integration of high-level alarms, auxiliary controls, unit mimic display, and safety parameter display system (SPDS) functions, into a wide display panel (WDP) used by the entire control room operating crew.

g) A control room layout (achieved through automation, distributed control technology, control building arrangement, and location of I&C panels) that significantly reduces personnel traffic and the need for non-operating personnel to access the main control room.

None of the above features and characteristics change the fundamental roles and responsibilities of the operators with respect to plant safety. Operators retain ultimate authority and decision-making responsibility. Operators are informed, involved, trained to understand the automation, and able to accomplish tasks in a timely, reliable manner.

No change will be made to the PSAR as a result of the response to this question.

2. The User's Manual will include a full list of all annunciated equipment errors. Software will be maintained in accordance with software configuration management requirements and procedures established for

Lungmen I&C. Correction of software errors will therefore occur in accordance with change control processes prescribed by the software configuration management requirements and procedures (e.g., tracking and reporting software errors, traceability of changes, status of reviews and audits, collection and retention of design records, etc.).

Correctness of operation is confirmed through reviews and checks of unit-level and system-level operating procedures (normal, abnormal, emergency, alarm response, etc.), using operators and the Lungmen plant simulator, during the verification and validation phase of the overall Lungmen Man-Machine Interface System (M-MIS) design and implementation process. [Refer to PSAR Section 18.7.2.8 and Parts VII(1)(d)(iii) and VII(2)(c & d) of PSAR Table 18.7-1.]

No change will be made to the PSAR as a result of the response to this question.

3. The Lungmen I&C design, particularly the digital safety-related systems, includes diagnostic features to detect errors. Self-test results (e.g., test messages on local or portable display and printout devices) are available to assist diagnostics work of personnel performing surveillance, testing, and maintenance tasks in accordance with applicable procedures. Software error was addressed by several, previously submitted PSAR questions. Please refer to the following responses regarding software error.
   a) The response to Track Number 07-030 addresses software error detection that meets the intent of GDC 21, RG 1.118, and IEEE 338 (Periodic Surveillance Testing).
   b) The response to Track Number 07-028 addressed meeting the intent of RG 1.53, IEEE 379, IEEE 603 Single Failure Criteria, and BTP-19 (Defense-in-Depth and Diversity) in protecting against interaction-induced, common mode failures in software-controlled systems.
   c) The response to Track Number 07-027 explains how the Lungmen I&C meets the intent of IEEE 603, Section 5.9 (Control of Access) in protecting itself against malicious intrusion from a software standpoint (e.g., virus, logical time-bomb, network intrusion, etc.).
   d) The response to Track Number 07-014 addresses compliance with

the SRP Chapter 7 and IEEE 279 by summarizing PRA analysis covering all RPS and ESFAS components and functions (as part of SSLC). The analysis accounted for potential common-mode failures due to software errors.

Aiding operators in failure detection and diagnosis is not the principal objective of an operating procedure, especially a symptom-oriented procedure. The following information is available and presented to operators during the course of an abnormal or emergency condition.

1. Safety parameter displays (to determine safety status, decide if manual action is needed, monitor engineered safeguards and mitigation, and execute emergency procedures). These displays include for example:
   a) Trend plots and validated readouts of reactor power, reactor pressure, and vessel water level (for the RPV Control EOP)
   b) Dynamic limits (e.g., SRV tail pipe level limits) including 2-dimensional plots (e.g., suppression pool level versus reactor pressure)
   c) ECCS actuation status (as relevant to a given EOP)
   d) Valve status (e.g., containment isolation valves)
   e) Building radiation monitoring
   f) Presence (or absence) of system component lineups
   g) Presence (or absence) of component power (electrical, hydraulic)

2. Fixed-position alarm tiles on the Wide Display Panel (WDP). These alarm tiles annunciate the following:
   a) Entry conditions to symptomatic emergency operating procedures (EOPs)
   b) Events such as containment isolation (inclusive of MSIV closure), reactor scram, turbine trip, and generator trip. *[Note: These type of event-based alarms have been used with "first hit", "first out", "first event" or "first trip" alarm logic in some plants. For Lungmen operators, alarm chronology and sequence-of-events information for post-event analysis and diagnosis is available on the VDUs.]*
   c) The presence of abnormal conditions within an individual system

d) Other time-critical, plant-level conditions

3. Alarm Response Procedures (available on CRTs)

4. Automated monitoring of Plant Technical Specifications for violations of Limiting Conditions of Operation, including display of recovery actions (available on CRTs).

No change will be made to the PSAR as a result of the response to this question.

4. HFE guidelines and practices of the U.S. nuclear industry are being applied to Lungmen (e.g., NUREG-0700, Human-System Interface Design Review Guideline, June 1996). These U.S. guidelines and practices broadly accommodate the 5th percentile female to 95th percentile male range of the U.S. adult population with respect to anthropometrics, perceptual capabilities (sensory), and motor skills (movement, manipulation, physical strength, etc.). It is assumed that specific features and capabilities of the Lungmen operator population are within the range of features and capabilities associated with the U.S. HFE guidelines and practices. Credible data specific to the Lungmen operator population is accounted for in the HFE design of Lungmen when such data is available. As noted in PSAR Section 18.4.2.2 (Main Control Console), panels and consoles in the MCR will be designed with anthropometric consideration of the user population applicable to Lungmen NPS.

   No change will be made to the PSAR as a result of the response to this question.

5. This question is similar to Track Number 07-023. The response to Track Number 07-023 stated that proven technologies, appropriate human factors principles, and human-centered automation collectively accounted for human cognitive strengths and weaknesses which were expressly included in the Lungmen NPS function allocation (automation) philosophy and criteria. The response to Track Number 07-023 summarized the human-centered automation approach for Lungmen as follows:

- The operator retains ultimate authority and decision-making responsibility
- The operator remains involved and is able to accomplish tasks within time, performance, and workload criteria
- The operator is informed and able to anticipate problems
- The operator understands the automation and can manage task support resources

Track Number 07-023 also provided a table of design features that ensured operator response and incident handling capabilities. Please refer to Track Number 07-023 for further details.

No change will be made to the PSAR as a result of the response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 18-004

PSAR Sections: Ch. 18

Question Date: April 5, 1998

PSAR Question:

1. NUREG-0711 2.4.1 (3) stated that human factors engineering program is applicable to main control room, remote shutdown facility, local control stations, technical support center and emergency operations facility but this chapter specified that human factor engineering program is only applicable to the first three. Please clarify.

2. Table 18.7-1 on members of human factors engineering design team does not include members with systems safety engineering, maintainability/inspectability engineering and reliability/availability engineering expertise as required by NUREG-0711, App. A. Please clarify.

3. Table 18.7-1(I)(4)(f) human factors qualification requirement is not consistent with NUREG-0711, App.A(6). Please clarify.

Response:

1. The human factors engineering program is also applicable to the technical support center and emergency operations facility as stated in Part 2.4.1(3) of NUREG-0711.

   The following sentence will be inserted immediately after the second sentence in PSAR Section 18.1 (Introduction):

   The Main Control Room (MCR), the Remote Shutdown System (RSD), the Technical Support Center (TSC), the Emergency Operations Facility (EOF), and local control stations are addressed by the HFE program.

2. Systems safety engineering, maintainability/inspectability engineering,

and reliability/availability engineering expertise are not absent from the overall M-MIS design and implementation process for Lungmen NPS. Some clarification is needed however. The composition of the "human factors engineering design team" (as it is called in NUREG-0711) for Lungmen is based on the same composition defined for the certified ABWR design. The USNRC, during its review of the HFE program plan for the ABWR, recognized the absence of systems safety engineering, maintainability/inspectability engineering, and reliability/availability engineering expertise. However, the USRNC found GE's composition acceptable, because the USNRC recognized that these particular areas of engineering expertise were applicable to the HSI design rather than the other HFE elements of the overall design and implementation process (Reference: Section 18.9.2.2.1 of NUREG-1503, USNRC Final Safety Evaluation Report for the ABWR, July 1994). System safety engineering, maintainability/inspectability engineering, and reliability/availability engineering expertise are included in the HSI design as part of the M-MIS design implementation plan for Lungmen NPS. Furthermore, system safety engineering is applied on an as-needed basis (i.e., safety engineers are not full permanent members of the HSI design team).

The following will be inserted as a note at the very end of Part I of PSAR Table 18.7-1 on page 18.7-7.

The composition of the "human factors engineering design team" (as it is called in NUREG-0711) for Lungmen is based on the same composition defined for the certified U.S. ABWR design. The USNRC, during its review of the HFE program plan for the ABWR, recognized the absence of systems safety engineering, maintainability/inspectability engineering, and reliability/availability engineering expertise. However, the USRNC found the composition acceptable, because the USNRC recognized that these particular areas of engineering expertise were applicable to the HSI design rather than the other HFE elements of the overall design and implementation process (Reference: Section 18.9.2.2.1 of NUREG-1503, USNRC Final Safety Evaluation Report for the ABWR, July 1994). System safety engineering, maintainability/inspectability engineering, and reliability/availability engineering expertise are included in the HSI design as part of the M-MIS design implementation plan for Lungmen NPS. Furthermore, system safety engineering is applied on an as-needed basis (i.e., safety engineers are not full permanent members of the HSI design team).

3. The text in question is slightly different from the text in NUREG-0711, but GE does not believe the difference constitutes an inconsistency with NUREG-0711 for the following reason. The current wording in PSAR Table 18.7-1, Part (I)(4)(f) is identical to GE's wording in ABWR SSAR Table 18E-1, Part (I)(4)(f) and the USNRC's wording in Appendix J of NUREG-1503 (USNRC FSER for the ABWR, July 1994). Furthermore, Part (I)(1) of PSAR Table 18.7-1 states that the composition of the M-MIS Design Team shall include, <u>as a minimum</u>, the technical skills presented in Part (I)(4). NUREG-0711, although also published July 1994, included some changes in its Appendix A (HFE Design Team Composition) wording as compared to the wording in the ABWR SSAR and the USNRC's FSER. The following identifies (in underline) pertinent differences in wording for the PSAR text in question:

Current PSAR Table 18.7-1, Part (I)(4)(f) Wording	NUREG-0711, Appendix A(6) Wording
Bachelor of Science degree in human factors engineering, engineering psychology or related science, and four years cumulative experience related to the human factors aspects of human-computer interfaces. Qualifying experience shall include <u>experience in at least two of the following human factors related activities;</u> design, development, and test and evaluation, and four years cumulative experience related to the human factors <u>field of ergonomics</u>. Again, qualifying experience shall include experience in at least two of the following areas of human factors activities; design, development, and test and evaluation.	Bachelor degree in Human Factors Engineering, Engineering Psychology or related science 4 years of cumulative experience related to the human factors aspects of human-computer interfaces. Qualifying experience should include <u>at least the following activities within the context of large-scale human-machine systems (e.g., process control):</u> design, development, and test and evaluation 4 years of cumulative experience related to the human factors <u>aspects of workplace design</u>. Qualifying experience should include at least two of the following activities; design, development, and test and evaluation.

PSAR Table 18.7-1 was not intended to duplicate, word for word, portions of applicable documents such as NUREG-0711. The incorporation of HFE principles into all phases of the Lungmen HSI design will be consistent with NUREG-0711 as noted in the introduction to PSAR Chapter 18.

No change will be made to the PSAR as a result of the response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 18-005

PSAR Sections: Ch. 18

Question Date: May 15, 1998

PSAR Question:

When systems operate normally, it becomes a question of how to keep a minimum work load for the operator to make sure he will maintain vigilance. This question involves rationalization of the work load design of digital I&C.

Response:

A workload (both physical and mental) that is within the operator's capabilities, and involves tasks and skills viewed positively by the operator, is just one of many contributors to vigilance (i.e., attentiveness, alertness, involvement). The HFE program addresses proper workload and vigilance which are central themes of human-centered automation for Lungmen to ensure operators remain involved, informed, and understand the automation. HFE principles and guidelines (most of which are embodied in criteria for function allocation, task analysis, and human-system interface design) are the rationale (i.e., the basis) for operator workload. One objective of the HFE principles and guidelines is to assure acceptable human performance under normal operating conditions. Validity of the HFE principles and guidelines, as applied to the design, is ultimately tested and proven as part of the Human Factors Verification and Validation phase of the HFE program. Designing for acceptable human performance, specifically vigilance under normal operating conditions, requires consideration of countermeasures against symptoms, and possible causes, of poor vigilance such as those listed in the table on the following page.

Previous responses to Question 7-023 (Batch 18) and Question 3 of 18-002 (Batch 8) also addressed workload, vigilance, situation awareness, human-centered automation, and design features (countermeasures). Parts of Question 7-023 closely related to countermeasures against poor vigilance are summarized below for convenience:

Features of the Lungmen Human-System Interface design

- Provisions for "hands on" operation and staying involved ("in the loop")

- Fixed-position switches
- Capability to conduct operations in a manual mode
- Capability to assume manual control by normal procedural methods and whenever operators elect to do so at their discretion
- Hold (break) points for automated unit startup and shutdown sequences

- Displa of automation information (actions taken, in progress, and pending)
- Teamwork an crew interaction (joint monitoring, sharing of information, task delegation, notification of key actions taken at control panels).

A training program tha emphasizes

- Understanding wat the automation does (both well and not well) and what the automation does not do
- Use of the trainng simulator to test for overdependence/overreliance on automation
- Motivatingoperators to learn by giving them the opportunities to experiment and learning from potential mistakes

Symptom of Poor Vigilance	Possible Cause
Somnolence (i.e., sleepy, sluggish, drowsy)	• A workload that is insufficient, overly simple, or highly inactive
Stress and/or Fatigue	• Environmental conditions (temperature, ambient noise levels, lighting, etc.) • Workplace/workstation designs with poor ergonomics
Complacency	• Boredom / Disinterest in work • Automation has limited the extent of manual, direct (ands on") operation. Tasks are predominantly passive rather than active (i.e., monitoring, acknowledging, confirming, permissives, etc.). • Over-dependence and trust on automation and ultra-quality systems (high reliability, availability, repeatability, etc.)
Neglect	• "Nuisance" indications (e.g., frequent expected alarms) • Relatively long waiting period for reaction/response from the automated system
Poor awareness and/or intuitive response	• Distraction by whatever competes for the operator attention • Lack of opportunity to apply skills and techniques acquired from prior on-the-job experience • Challenge and problem-solving (abnormal conditions, emergencies, etc.) are rarely encountered/experienced • Deficient training and/or procedures

	• Actions and status of the automation are difficult to understand • Inadequate time for operator interpretation, evaluation, and response.
Combinations of the above	[Factors that are beyond the control of the designer such as physical health or personal matters negatively affecting attitude.]

No change to the PSAR will be made as a result of the response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

原能會審查意見(ROCAEC Review Comments):

請參考審查問題07-023之答覆方式列表說明龍門計畫解決Poor Vigilance問題之Countermeasure。

台電澄清說明(Further Clarification):

龍門計畫解決 Poor Vigilance 問題之 Countermeasure 列表如下：

Symptom of Poor Vigilance	Possible cause	Countermeasure
Somnolence (i.e., sleepy, sluggish, drowsy)	● A walkload that is insufficient, overly simple, or highly inactive	● Teamwork and crew interaction (Joint monitoring, Sharing of information, task delegation, notification of key actions taken at control panels).
Stress and/or Fatigue	● Environmental conditions (temperature, ambient noise levels, lighting, etc.) ● Workplace/workstation designs with poor ergonomics	● HSI design meet HFE requirements ● Shifting organization ● Related training course ● Human Performance Enhancement System ● Function allocation to machine ● Social contact at work
Complacency	● Boredom / Disinterest in work ● Automation has limited the extent of manual, direct ("hands on") operation Tasks are predominantly passive rather than active (i.e., monitoring, acknowledging, confirming, permissives, etc.). ● Over-dependence and trust on automation and ultra-quality systems (high reliability, availability, repeatability, etc.)	● Teamwork and crew interaction (Joint monitoring, Sharing of information, task delegation, notification of key actions taken at control panels). ● Provisions for "hands on" operation and staying involved (" in the loop ") - Fixed-position switches - Capability to conduct operation in a manual mode - Capability to assume manual control by normal procedural methods and whenever operators elect to do so at their discretion

Symptom of Poor Vigilance	Possible cause	Countermeasure
		- Hold(break) points for automated unit startup and shutdown sequences ● A training program that emphasizes - Understanding what the automation does (both well and not well) and what the automation does not to do - Use of the training simulator to test for overdependence / overreliance on automation - Motivating operators to learn by giving them the opportunities to experiment and learning from potential mistakes
Neglect	● "Nuisance" indications (e.g., frequent expected alarms) ● Relatively long waiting period for reaction / response from the automated system	● Prevent " nuisance alarm ", " nuisance indication " ● HSI response time requirements ● " alert ", " prompt ", " remind ",.........design ● Training
Poor awareness and/or intuitive response	● Distraction by whatever competes for the operator's attention ● Lack of opportunity to apply skills and techniques acquired from prior on-the-job experience ● Challenge and problem-solving (abnormal conditions, emergencies, etc.) are rarely encountered / experienced ● Deficient training and / or procedures ● Actions and status of the automation are difficult to	● Teamwork and crew interaction (Joint monitoring, Sharing of information, task delegation, notification of key actions taken at control panels). ● Provisions for "hands on" operation and staying involved ("in the loop) - Fixed-position switches - Capability to conduct operation in a manual mode - Capability to assume manual control by normal procedural methods and whenever

Symptom of Poor Vigilance	Possible cause	Countermeasure
	understand ● Inadequate time for operator interpretation, evaluation, and response	operators elect to do so at their discretion - Hold(break) points for automated unit startup and shutdown sequences ● Display of automation information (action taken, in progress, and pending) ● A training program that emphasizes - Understanding what the automation does (both well and not well) and what the automation does not to do - Use of the training simulator to test for overdependence / overreliance on automation - Motivating operators to learn by giving them the opportunities to experiment and learning from potential mistakes
Combinations of the above	[Factors that are beyond the control of the designer such as physical health or personal matters negatively affecting attitude.]	● Administration management ● Safety culture ● Award and penalty ● Physiological and psychological aid

原能會進一步審查意見(ROCAEC Review Comments):

一、龍門儀控設計對安全有關系統提供何種自動控制功能？

二、系統如何判定何時轉為手動？如何防制錯判？

三、無論自動或手動，系統是否提供每一步驟操作結果？運轉員是否可維持狀況掌握（Situation Awareness）？

台電再澄清說明(Further Clarification):

一、 PGCS 為最上層的電廠自動化控制器，PGCS 為非安全有關系統不得直接控制安

全系統，當需要安全系統改變狀態時(僅為最起碼的手動動作，如轉變反應器運轉模式開關位置)，PGCS 自動提供提示，運轉員手動完成操作後，需壓按一個段落點控制按鈕告訴 PGCS 本步驟已完成，可繼續後面的步驟。至於安全系統之自動動作設計方式則與既有電廠一致，不屬本自動化範圍。

二、PGCS 有一運轉模式選擇開關，運轉員可選擇自動、半自動、或手動模式，任何時間運轉員可轉為手動，PGCS 會自動檢查、比對機組正常運轉狀態變化，一旦機組發生異常狀況，PGCS 立即暫停自動模式(詳參附件 PGCS HSS 資料及圖)，而各別系統及設備按其設計來反應處理當時之狀況。若電腦故障，系統自動脫離 PGCS 自動控制。故防制錯判係靠遵照龍門儀控設計準則進行相關設計，確實提升硬、軟體可靠度、採用分散式 CPU 配置、執行程式驗證與確認、完整的功能測試、提升信號品質、信號多重性、電腦自動定期測試與校正、及整廠預防電磁干擾等使系統能提供正確運轉資訊，如此系統與運轉員均不致誤判。

三、龍門人-系統界面設計需進行功能分析、功能配置、作業分析後再據以設計人-系統界面，故全部作業所需之顯示、控制、警報均將具備，而 PGCS 對每一步驟操作結果均由取信器回饋，並且提供以程式合成的有意義的狀況顯示。維持運轉員狀況掌握為龍門人-系統界面設計眾多考慮因素之一，並將在最終接收試驗時以程序書由運轉人員驗證與確認，故控制室未來所提供資訊將足夠，運轉員應可隨時維持狀況掌握。

附送 GE PDM 內有關章節供參考。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-001

PSAR Section: 19. 3

Question Date: January 14, 1998

PSAR Question:

1. The axis scaling and curve labeling in Figure 19.3-1a have caused confusion, please re-plot it.

2. What is the dash-line curve in Figure 19.3-2d? Is it the same as lower drywell temperature shown in Figure 19.3-2c?

PSAR Response:

1. Figure 19.3-1a is meant to show only the drywell pressure behavior for the analyzed station blackout accident scenario. A printing error apparently resulted in the overlay of the labels of Figure 19.3-1b on Figure 19.3-1a leading to the regrettable confusion. The correct figure is attached.

2. The two cited curves represent different materials in the drywell: The dotted curve in Figure 19.3-2d shows the temperature of the corium dropped in the lower drywell after vessel failure, and the lower drywell temperature dotted curve in Figure 19.3-2c presents the lower drywell gas temperature. The two curves peak at 23.5 hours when the lower drywell gas temperature reaches 533° K causing the passive flooder to open and cool the drywell gas and the corium on the drywell floor.

No PSAR revision is proposed in response to Part 2 of this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-002

PSAR Section: 19.4

Question Date: January 14, 1998

PSAR Question:

1. In Section 19.4.3.3.1, the maximum drywell pressure after severe accident is approximate 0.79 MPa, while the pressure is 0.7 MPa in AJ.10.1, please adjust them.
2. The RPV and Containment failure probabilities in Section 19.4.3.3.2.3 and AJ.10.4.2 are inconsistent, please justify.

Response:

1. The approximate maximum drywell pressure of 0.7 MPa quoted in AJ.10.1 is incorrect and will be corrected to the value of 0.79 MPa of Section 19.4.3.3.1 which is more appropriate for high pressure core melt ejection accident sequences. As indicated in Section 19.4.3.7.2, MAAP analysis of the dominant sequences involving RPV failure at high pressure results in the bounding drywell pressure shown in Figure 19.4-27a. As seen in that figure, the drywell pressure peaks at approximately 0.79 MPa.

2. The RPV and containment failure probabilities of Section AJ.10.4.2 are incorrect. An inspection of Section AJ.10 revealed that the whole section has been erroneously copied from the file containing the Lungmen PRA in the original proposal to TPC. It is particularly noted that the proposal PRA estimated a relatively low CDF which resulted in the relatively large (0.562) conditional probability of high RPV pressure at vessel failure. This is the main reason for the increase of the conditional containment failure probability due to DCH from 1.0E-3 value reported in the SSAR[1] to the value of 3.0E-3 which is based on the original Lungmen proposal PRA assumptions.

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[1]The SSAR value of 1.E-3 is an approximation of the value of 1.37E03 shown in the attached Table 19-002-1.

The Lungmen ABWR design is developed to meet requirements (TPC Requirements Document) which are very similar to those used for the certified ABWR (Utilities Requirements Document). These similarities justified the use of SSAR severe accident analysis for the Lungmen NPS as indicated in the introduction of Attachment AJ of PSAR Appendix A. Consistent with this position, differences between Appendix AJ and SSAR Chapter 19, Appendix EA, have been identified and discrepancies have been corrected. The affected pages are: AJ.10-1, 3, 26, 27, 33 through 35, and 37. Markups of these pages are attached.

In view of the above finding, Section 19.4.3.3 was also inspected for possible inconsistencies. Unfortunately, errors were also found in Figures 19.4-11 and Figures 19.4-13 through 19.4-15. These errors were also corrected. Markups of these pages are attached.

It is recognized that no seismic PRA is included in the SSAR and that the SSAR used a 0.3g SSE which is different from the Lungmen SSE of 0.4g. To ensure that the Lungmen DCH-induced containment failure probability is low under severe accidents initiated by internal and seismic events, an analysis was performed using the internal events and seismic PRA core meltdown sequences reported in Appendix A, Attachment AB (Internal Events PRA) and AC (Seismic PRA). The results of the above analysis is shown in the attached Table 19.002-1 and is summarized below. The table is based on the seismic accident sequences of Table 19.002-2. Table 19.002-2 contains CDF for all accident sequences identified in the event trees of Figure AC.6-1, Attachment AC of the PSAR Appendix A. The table identifies sequences leading to high RPV pressure at vessel breach and the pre-existing containment pressure. These pressures were used in the estimation of the DCH-induced conditional containment failure probability. Notice that the table identifies sequences where the Emergency Operating Procedure (EOP) may lead to depressuriztion before RPV breach. This is more realistic than the conservative seismic analysis presented in Table AC.10-2, where no EOP credit was taken for these sequences.

The estimated frequency of the Lungmen high RPV pressure sequences is 3.68E-8 per year for internal initiating events. With a total internal events core damage frequency of 2.25E-7 per year, the conditional probability of high RPV pressure on vessel failure is 3.68E-8 / 2.25E-7 = 0.164. For the seismic events, the estimated frequency of the high RPV pressure sequences is 5.52E-7 per year. With a total seismic core damage frequency of 3.14E-6 per year, the conditional probability of high RPV pressure on vessel failure is 5.52E-7 / 3.14E-6 = 0.176. These values are lower than the SSAR value of 0.273. Using Lungmen-specific probability distribution for pre-existing containment pressure (see attached table) leads to a conditional probability of DCH-induced containment failure of ~ 8.3E-4 for internal events and 1.3E-3 for seismic events. Since seismic events dominate the Lungmen risk, the average DCH-induced containment failure for all initiating events is expected to be similar to the seismic value of 1.3E-3. These values are comparable to the SSAR DCH-induced containment failure probability value of 1.0E-3. This provides an assurance of the Lungmen robustness against DCH containment failure.

PSAR Sections 19.4.3.3.2.3, 19.4.3.3.2.4, AJ.10.4.2, and AJ.4.2.1 have been revised to reflect the results of the above Lungmen-specific analysis. Markups of these sections are attached.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-003

PSAR Section: Ch 19.4.3.1

Question Date: November 24, 1997

PSAR Question:

1. What is the basis for the assumption that all heat generated in the Metal-Water reaction are absorbed in the suppression pool ? Is this assumption conservative enough for estimating the containment pressure?
2. The analysis results in this section only showed that the peak pressure will be 0.618 MPa. Please also add the changes in parameters for Drywell Pressure, Drywell Temperature, Hydrogen Mass, Reaction Heat, Wetwell Temperature and Wetwell Pressure, etc.

PSAR Response:

1. The cited assumption is based on the basic design of a pressure suppression containment and is consistent with licensing basis calculations. The purpose of this section is to demonstrate compliance with 10CFR50.34f. Assuming the energy is transferred to the suppression pool is conservative because the heatup of the suppression pool controls the overall containment pressurization. Any energy absorbed in-vessel or containment structures will reduce containment pressurization. Any energy absorbed by the gases in the vessel or drywell will initially result in an increase in drywell pressure. However, this will cause gas flow into the suppression pool via SRVs or the containment vent system and the energy will be transferred to the pool. An estimate of the containment pressure using MAAP analysis of a 100% active fuel cladding-water reaction scenario indicates that the above modeling assumption leads to a reasonable containment pressure estimate. The MAAP analysis does not use licensing basis assumptions. This analysis is described below.
   Section 19.4.3.7.1.1 presents MAAP analysis of an accident scenario that leads to 100% active fuel cladding-water reaction. As described in that section, the scenario starts by an isolation event, and contains

several intermittent operations of the ECCS and conservative assumptions to ensure that 100% of the cladding reacts. As seen in Figure 19.4-25a, the resulting drywell pressure is almost constant at 0.38 MPa.

2. Two models have been used for the 100% metal water reaction analysis. The first model is the simplified model described in Section 19.4.3.1 to ensure that the containment pressure is within the Level C pressure limit under a combined design basis LOCA and an assumed 100% Zr-water reaction. This analysis was a lumped analysis where the reaction heat and hydrogen mass corresponding to the Zr mass involved in the reaction were estimated. All the hydrogen generated was then allowed to distribute instantly in the containment atmosphere, and the reaction heat was deposited in the suppression pool. The second model involves a hypothetical scenario that was developed for analysis by the MAAP-ABWR code to ensure interaction of 100% of the active fuel cladding. The MAAP-ABWR analysis is mechanistic and time dependent, although the scenario used to generate the 100% of the active cladding is extremely unlikely. The analysis accounts for the interaction rate, reaction heat generated, and partitioning of the reaction heat between the generated hydrogen, steam, and reactor core components and structures through proper thermal hydraulic analysis. Section 19.4.3.7.1.1 provides the details of the scenario used in the MAAP analysis and the results of this analysis.
   The MAAP-ABWR time-dependent drywell and wetwell pressure and temperature estimates for the above 100% metal-water reaction scenario are presented in Figures 19.4-25a through 19.4-25e. The total hydrogen mass corresponding to the active fuel cladding oxidation to ZrO2 is 1,600 Kg and the corresponding reaction heat is 260,200 MJ. The hydrogen and heat estimates are based on a mass of 38,410 Kg of the active fuel cladding. These estimate are based on the SSAR fuel. The Lungmen GE-12 fuel will produce 3% less hydrogen and heat energy than those of the SSAR. Therefore, the SSAR results are conservative.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

ROCAEC Review Comments:

(1) As indicated in the responses to 19-003 question 1, any energy absorbed by the gas in the drywell will result in an increase in the drywell pressure. Thus, the assumption that all heat generated in the MWR (Metal Water Reaction) are absorbed in the suppression pool may underestimate the peak pressure of the drywell. Please assumption that all heat generated in the MWR (Metal Water Reaction) are absorbed in the suppression pool may underestimate the peak pressure of the drywell. Please clarify.

(2) Responses to question 2 do not answer the question. We understand there are two models for the analysis. We like to have the results for simplified model analysis. Please provide the response of the siestion 2 do not answer thhe question. We understand there are two models for the analysis. We like to have the results for simplified model analysis. Please provide the response of the simplified model (with peak pressure of 0.618 MPa), including the wetwell pressure, wetwell temperature, drywell pressure, drywell temperature, hydrogen mass, and reaction heat.

Further Clarification:

(1) As indicated in Section 19.4.3.1.2, page 19.4-7, the simple model used in the 100% MWR analysis did not take credit for heat transfer to the drywell or wetwell heat sinks. Moreover, the model aanalylysis did not take credit for heat transfer to the drywell or wetwell heat sinks. Moreover, the model assumes that deposition of all of the MWR energy in the suppression pool and hydrogen addition to the drywell and wetwell atmosphere occur instantaneously at the time of peak LOCA pressure (a few seconds after the break) although the MWR reaction is expected to occur much later (after the containment pressure has dropped significantly). These assumptions overestimate the pressure increase from the MWR and more than offset the effect of ignoring the MWR energy deposited in the containment gases.

(2) The results of the simple MWR analysis are: wetwell pressure 0.618

MPa, wetwell temperature 140 °C,. drywell pressure 0.618 MPa, drywell temperature 140 °C, hydrogen mass 1684 Kg, and reaction heat 2.6E5 MJ.

ROCAEC Review Comments:

As shown in response to Comment No. 1 (Track No. 19-003), the following explanations are given.
The simplified model is a lumped, time-independent model used to calculate an upper bound of the peak pressure. The simplified model assumes that deposition of all the MWR energy in the suppression pool and hydrogen addition to the drywell and wetwell atmosphere occur instantaneously at the same time of peak LOCA pressure. The results indicate the same pressure (0.618 MPa) and temperature (140 °C) for both of the wetwell and drywell.

Questions:

1) Please explain how the peak LOCA pressure was calculated? From MAAP-ABWR analysis? What is the peak pressure of the drywell and wetwell?
2) The lumped, time-independent simple model seems to calculate the final equilibrium pressure of the lumped drywell and wetwell system, instead of the peak pressure of the drywell. Please explain.
3) With the lumped, time-independent simple model, what is the difference in peak pressure with the assumption that all heat generated in the MWR is absorbed in the drywell?

Further Clarification:

1) The procedure for calculating the peak LOCA pressure is explained in PSAR Chapter 6, Section 6.2.1.1.3. The procedure provides design basis analysis which is more conservative than MAAP-ABWR mechanistic analysis. As discussed in Section 6.2.1.1.3, the feedwater pipe break design basis accident leads to the maximum drywell and wetwell pressures. As seen in PSAR Table 6.2-1, page 6.2-82, the drywell peak pressure is 268.7 kPaG and the maximum wetwell peak pressure is 179.5 kPaG. Figure 6.2-8a, page 6.2-154, shows that these maximum pressures occur in the early phase of the accident.

2) The simple model uses the above design-basis LOCA peak drywell pressure and superimposes on it the pressure increase due to hydrogen mass addition to the containment atmosphere and the metal-water reaction energy addition to the suppression pool. To calculate the above pressure increase, the model uses the design basis LOCA short term maximum drywell and suppression pool temperatures (140 °C and 60 °C respectively - See Figure 6.2-7, page 6.2-152) as the initial conditions. The model simplifies the analysis by combining the drywell and wetwell airspace as a single node with the same pressure and temperature as those of the drywell. The model further assumes instantaneous uniform mixing of the 100% MWR hydrogen mass in the drywell and suppression pool airspace, and instantaneous deposition of the MWR heat in the suppression pool water. This assumption is made because the containment sprays are assumed to be in operation in accordance with the 100% MWR Licensing Basis for ABWR. Depositing the MWR heat energy in the suppression pool increases its temperature and consequently the steam vapor pressure. The steam vapor pressure increase is added to the pressure increase due to hydrogen mass addition and the peak design basis LOCA drywell pressure to obtain the total drywell pressure. As indicated by the above summary, the drywell pressure estimated by the simple model is not the final equilibrium pressure (since it includes the instantaneous peak LOCA pressure).

3) The simple model assumes that the operator initiates the containment sprays to reduce the containment pressure. The sprayed water will remove the heat from the containment atmosphere and deposit it in the suppression pool. The assumption has been used in the ABWR certification and accepted by the USNRC for the 100% MWR containment analysis.
Since hydrogen generation is a time-dependent process and pressurization of the drywell forces flow to the wetwell, assuming that all the heat is deposited in the drywell (instantaneously as required by the simple model) is unrealistically conservative. Use of the simple model in this unrealistic fashion results in a drywell temperature >22,000 °C. This, of course, is an unrealistic result and reflects the limitations of the simple model. The proposed assumption of depositing all the MWR reaction heat in the drywell can only be handled with a

more complex model that accounts for time dependence of the MWR energy and hydrogen release, drywell/wetwell pressure difference and gas flow, and temperature-dependent specific heats and heat transfer. The MAAP-ABWR code has the capability to take these factors into account. As indicated in the original response to this question, the MAAP-ABWR mechanistic analysis led to a containment peak pressure lower than that of the simple model.

ROCAEC Review Comments:

Please commitment to perform MAAP analysis of containment pressure in FSAR.

Further Clarification:

Time-dependent analysis of the combined LOCA and 100% MWR accident will be performed for the FSAR. One of two options will be considered for this analysis: using the MAAP-ABWR code, or improving the simple time-independent model used in the PSAR to allow time-dependent analysis. GE likes to keep these two options open for now. A commitment to use the MAAP-ABWR option can not be made at this time due to the following considerations.

A key difficulty with MAAP-ABWR analysis is defining an accident scenario that can lead mechanistically to 100% metal water reaction (MWR) in the RPV. All mechanistic scenarios analyzed for the SSAR led to only a few % MWR in the RPV. A fictitious case was run for the SSAR to simulate a TMI-type accident with 100% MWR. The case is reported in PSAR Section 19.4.3.7.1.1, page 19.4-74. The case involves an isolation event and not a LOCA. The analysis was forced to produce 100% MWR by assuming ECCS cycling and changing the fuel rod geometry to increase the MWR rate. The analysis was not easy and took several iterations and more than 14 hours on the computer without interruption to maintain the RPV water level in the right range to obtain the 100% MWR.

Doing an analysis similar to the above for LOCA is going to be tedious and its convergence within a realistic computation time is not certain. For this reason, GE likes to keep the option of improving the simple time-independent model used in the SSAR by making it able to account for the time dependence of the hydrogen generation, reaction heat, drywell pressure, and other key variables. Naturally, insights from available MAAP-ABWR analysis will be used to define input and modeling assumptions for the improved simple model if it is chosen for analysis.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-004

PSAR Section: Ch 19.4.3.2

Question Date: November 24, 1997

PSAR Question:

1. Please explain the difference between the calculations of MAAP-3B and MAAP-4B for corrosion of concrete foundation.

2. ANL successfully completed the M3B Debris Coolability experiment in 1997 which helped the explanation of the debris cooling phenomenon. It should also supplement the results of the M1B experiment in 1991 of this section. Please explain if it should be included in the relevant discussions ?

PSAR Response:

1. There are two main differences between the two MAAP versions in the way they analyze concrete basemat erosion:
   i) In MAAP3B, the gas liberated from the sidewalls is assumed to bypass the debris without thermal or chemical interaction. In MAAP4, a user option is provided to allow these interactions, with a default value that allows complete reaction of the released gases. The effect of this difference is substantial for deep debris beds with significantly more concrete erosion estimated by MAAP4 than that estimated by MAAP3B. For shallow beds like the ones in the ABWR (due its large drywell floor area), the above difference is expected to have little impact on the results.
   ii) In MAAP3B versions issued after revision 6.06, radiation from the drywell debris is not accounted for. The MAAP-ABWR does not have this deficiency because it was developed from an earlier version.

2. The M3B experiment will be reviewed for applicability to the Lungmen ABWR and included in the FSAR if appropriate.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-005

PSAR Section: Ch 19.4.3.4

Question Date: November 24, 1997

PSAR Question:

1. In the discussion of Rupture Disk vs. Temperature effects, what accident sequence was based for the temperature range of 311 to 422K ?

2. In the discussion of Rupture Disk size, how was the Minimum Acceptable Flow of 28 kg/sec obtained ?

PSAR Response:

1. Severe accident sequences analyzed by MAAP3.0B-ABWR in Attachment AJ of the PSAR Appendix A that involve COPS operation show that the wetwell air space pressure reaches 0.72 MPa at temperatures less than or equal to 422 K. The range from 311 - 422 K covers the operating conditions from cold shutdown to peak accident temperatures. The discussion in Section 19.4.3.4 indicates that the COPS disks rupture pressure changes only by 2% of the nominal pressure of 0.72 MPa (366 K).

2. The above minimum flow rate was determined by the limiting ATWS analysis discussed in the first paragraph of page 19.4-43, which indicates that the flow of 28 kg/sec is required to keep the containment pressure below service level C.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-006

PSAR Section: Ch 19.4.3.6

Question Date: November 24, 1997

PSAR Question:

1. In the analysis of fuel and coolant water reaction, experimental results were used (Vapor Explosions in a Stratified Geometry, K.H. Bang and M.L. Corradini, 1991) to deduce that there are three times margin for the reactor foundation to sustain the FCI produced pressure wave. But that experiment was done for Freon/Water and Liquid Nitrogen/Water Systems. Is it applicable to ABWR ? Please explain.

2. Recently, in the CSARP project, lots of experiments were done for FCI [such as KROTOS, FARO, ZREX(ANL), WFCI(University of Wisconsin)]. Can the conclusions from those experiments be applied to Lungmen ? Please explain.

3. During Vessel Breach, if Cavity is flooded then there is possibility of steam explosion. What is the basis for the reasoning that the possibility of Cavity flooding being too low to be taken into account ? What steps have been taken to prevent it ? If steam explosion does occur, can the containment take the impact ?

PSAR Response:

1. As stated in p. 19.4-58, tests using simulant materials are not directly applicable to the reactor condition, but they are help to understand the underlying physics of FCI. A review of the prototypical MACE and WETCOR tests (p. 19.4-59) shows no energetic FCI for stratified geometry. The reasons for using the experimental results of K.H. Bang and M.L. Corradini, 1991 are that: a) They are the only tests reporting energetic FCI in stratified geometry, b) They provide estimates of the depth of liquid-liquid mixing which is necessary for FCI explosions and show that the depth is very small (<1 cm) over the range of experimental conditions used. The depth of 1 cm was used as an

upper bound of mixing depth to calculate the FCI corium mass for the ABWR. The factor of 3 capability of the pedestal provides a margin of safety against uncertainties in the above calculations.

2. The tests conducted in these facilities aim at better understanding of separate phenomena effects and for validation of computer codes. However, the tests and their results will be followed and evaluated for applicability to the Lungmen ABWR.

3. The USNRC suggests, as a containment performance criterion, a conditional containment failure probability < 0.1 given the spectrum of accidents leading to severe core damage. Consistent with this criterion, a guideline used to simplify the ABWR analysis was to ignore sequences leading to early containment failure if their CDF is < 0.01*Total CDF. Analysis in 19.4.3.6.2.1, page 19.4-58, shows that the CDF of sequences that lead to water in the lower drywell before RPV failure is less than 0.003 *Total CDF, which justified ignoring further analysis of ex-vessel steam explosion.

   ABWR features that prevent water from flooding the reactor cavity before RPV failure are discussed in Section 19.4.3.6.2. It should also be noted that the ABWR has intrinsic and passive features that limit the mass of corium that could participate in a violent FCI. These features include RPV depressurization, lower RPV penetrations and below-RPV structures. These features slow down the corium injected in the reactor cavity, cause incoherence in the arrival of corium debris to the lower drywell floor, and cool the corium. This limits the potential for an energetic FCI causing containment failure.

   The above ABWR provisions provide assurance that energetic FCI will not be a dominant contributor to the risk or significantly impact the containment performance as measured by the above USNRC containment criterion.

No PSAR revision is proposed in response to this question.

ROCAEC Review Comments:

Please provide information of the guideline that ignores sequences leading to early containment failure if their CDF is < 0.01* total CDF. Is this guideline approved by USNRC?

Further Clarification:

The above guideline sets an upper bound for the truncation of analysis in case this is necessary to focus on areas of risk significance and avoid excessive detailed analysis of relatively small risk contributors. It amounts to ensuring that at least 90% of the risk associated with early containment failure (which is a fraction of the total risk) is analyzed in adequate detail. The criterion has not been proposed or approved by the USNRC. However, the FSER states in the second paragraph of Page 19-63 that "The staff believes that the low likelihood (0.003) of a flooded lower drywell at the time of reactor vessel failure provides a sufficient basis to conclude that the probability of an ex-vessel steam explosion has been reduced to an acceptably low value and is therefore acceptable."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-007

PSAR Section: Ch 19.4.3.8

Question Date: November 24, 1997

PSAR Question:

1. Please explain how the channel width and number are determined ? (page 19.4-87, starting line 15)

2. Please explain the basis and consideration of the assumption that the total channel width is 2m (Page 19.4-104, 12th line from the bottom).

PSAR Response:

1. The channel size was selected to provide a flow area equal to the area of the 10 cm diameter pipe connected to the sump draining pump. To enhance debris freezing within the channel, the channel height was chosen as small as possible (1 cm). From the calculated flow area (~ 80 cm^2) and channel height, a total width of about 0.8 m was calculated. To ensure a uniform channel height, the total width was divided into four channels, which led to a single channel width of 0.2 m.

2. The 2 meter channel width was conservatively assumed in the SSAR to estimate an upper bound for the amount of corium ingression into the sump under very unlikely bounding corium superheat conditions (Scenario M). This resulted in the estimated 0.006 m^3 of debris ingression and the corresponding 0.2 cm debris depth in the sump. For the actual total channel width of 0.8 m indicated in the response to question 1 above, the upper bound of the corium ingress and depth will be only 40% of the above values, i.e., 0.0024 m^3 and 0.08 cm.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-008

PSAR Section: Ch 19.5

Question Date: November 24, 1997

PSAR Question:

The analysis tool is the MAAP3.0B (completed in January 1988) modified MAAP-ABWR. The new version MAAP 4.0 has been issued in May 1994 with improvements in Accident Management and ALWR model, Core Melt Progression and RPV failure model, In-vessel Cooling and containment model, etc. Is this new version MAAP 4.0 going to be used during FSAR ?

PSAR Response:

The MAAP3.0B-based MAAP-ABWR was developed for the analysis needed for certification of the Standard ABWR which was completed in 1994. Validity of the above analysis has been confirmed by the USNRC-sponsored analysis reported in Reference 19-008.1. The reference includes analysis of five of the SSAR accident sequences using the MELCOR 1.8.2 Code and compares the results to those of the MAAP-ABWR. The main conclusions of the above comparison are:

i) MAAP-ABWR and MELCOR produced similar time trends of key variables.
ii) MELCOR generally predicts later times for core uncovery and slower core damage progression than MAAP-ABWR.
iii) COPS disk rupture time and release fractions of radionuclides predicted by the two codes are comparable when debris quenching is included in MELCOR as it is in MAAP-ABWR.

Following the release of MAAP4.0, an evaluation of its new features was performed by the US Standard ABWR Project in 1994. It was noted that the MAAP4.0 new ALWR models have been developed to analyze passive features of the small LWRs such as the SBWRs, and that the code must be modified for application to the ABWR. It was also concluded that: a) numerical instabilities have been reported by some MAAP4.0 users, and b) for the ABWR design, the MAAP4.0 new features will not materially change the SSAR results (Please see related information in response to question 19-004.). Consequently, there was no compelling reason for the US Standard ABWR project to allocate the resources required for MAAP-ABWR code modification and the necessary validation and verification (V&V).

Over the past three years, the MAAP4.0 code has been increasingly used for Level-2 PRA analysis by USA utilities. The Spanish Nuclear Regulatory Commission (CSN) reported its intent to use the MAAP4.0 Accident Response System (MARS) for real-time severe accident tracking and management. Unfortunately, however, the MAAP Users Group have reported some errors discovered during their application of MAAP4.0. A serious error which is important for the issue of severe accident management has been reported in 1997. The error was discovered in a heat transfer subroutine which resulted in significant underestimation of the time to vessel failure. Although the error has been corrected by the MAAP4.0 developer, modifying the code for ABWR application and performing the necessary V&V remains a significant undertaking with questionable benefits.

Based on the above considerations, the Lungmen Project will continue using the USNRC validated MAAP3.0B-ABWR along the following lines:

i) To show that the USNRC-accepted and validated SSAR analysis adequately bounds the severe accidents identified for the Lungmen NPS
ii) To conduct further sensitivity analyses using the MAAP3.0B-ABWR to evaluate the effect of key uncertainties that may be identified from review of MAAP4.0 new features and reported analyses, and from on-going severe accident tests.

No PSAR revision is proposed in response to this question.

References:

1. L. N. Kmetyk, "MELCOR 1.8.2 Calculations of Selected Sequences for the ABWR," SAND94-0938, July 1994.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-009

PSAR Section: Ch 19.6

Question Date: November 24, 1997

PSAR Question:

1. Please explain what the ABWR Severe Accident Management (SAM) Strategies are ?
2. Ex-vessel Cooling is a main strategy to prevent Vessel Breach. Is it going to be abandoned for ABWR ? If yes, is there a replacement strategy ?
3. One of the major improvements of MAAP 4.0 is its added accident management capability. Please explain if MAAP 4.0 going to be used in FSAR for the analysis of this section.

PSAR Response:

1. The ABWR severe accident management strategies include those actions by the plant operator and local authorities aimed at reducing the public risk from such accidents should they occur. Risk reduction areas include:
   i) To terminate core damage
   ii) To ensure long term retention of a damaged core within the reactor pressure vessel
   iii) To ensure containment integrity
   iv) To minimize radioactive materials release from the secondary containment
   v) To minimize public exposure due to accidental release of radioactive material.

   Actions to accomplish the above safety objectives will utilize available safety-related and non-safety related equipment. Examples of such actions include alignment of an isolated water source or an electric power source. The specific actions to be taken depend on the specific accident scenario and availability of equipment that can accomplish the safety objective. As stated in Section 19.6, development of the Lungmen-specific risk management strategies will be based on insights from the Lungmen PRA and the identified strategies will be included in the FSAR.

1. Ex-vessel cooling has never been considered as a risk management strategy to prevent reactor vessel breach in the ABWR. It is recognized that such a strategy has been recommended as a last resort and under certain constrained conditions for conventional BWRs in Revision 4 of the BWR Owners Group EPGs. However, application of this strategy to the ABWR has been ruled out from the beginning because it was judged as risky with questionable reliability and benefits as illustrated below:

   i) Containment flooding compromises the containment integrity in two respects:
      a) It requires containment venting to prevent containment overpressurization as a result of the reduced gas space caused by water addition.
      b) It increases the chances of violent molten fuel coolant interaction which threatens the containment integrity.

   ii) Containment flooding must be quick enough to reach approximately the top of active fuel height and requires electric power which is not available under the risk important SBO conditions.

   iii) Applying the concept of ex-vessel cooling before vessel breach requires a significant R&D effort to establish an adequate level of understanding of the heat and mass transfer processes under prototypical conditions.

   iv) The ABWR safety philosophy is based on making the containment as fully independent fission product barrier as is reasonably possible by use of the preventive, passive measures against containment threats discussed in Chapter 19. These measures include:
      a) RPV depressurization by the ADS to prevent direct containment heating
      b) Minimizing the chances of having water in the lower drywell before vessel breach in order to avoid FCI-caused steam explosion
      c) Use of sacrificial basaltic concrete to minimize the impact of corium-concrete interaction

d) Use of the passive drywell flooder which is actuated after vessel breach to quench the released corium.

e) Holding the released fission products in the containment as long as possible while allowing containment pressure relief through the COPS after the contaminated containment atmosphere has been decontaminated by scrubbing in the suppression pool.

The above provisions have been accepted by the NRC as viable engineered provisions to protect the containment and minimize the impact of uncertainties in such phenomena as FCI-initiated steam explosion, which may occur in case of containment flooding.

1. Please see response to question 19-008.

No PSAR revision is proposed in response to this question.

ROCAEC Review Comments:

It seems unrealistic to analyze the modern ABWR severe accident management related work in the FSAR (long time from now) with an obsolete tool for the following reasons:

(1) MAAP3B does not have most of the capability of recovery actions in the severe accident management guideline (SAMG) analysis and so does the MELCOR.

(2) USNRC consider the SCDAP/RELAP5 as the tool for SAMG analysis. EPRI develops MAAP4.0 as the tool for SAMG analysis and consider MAAP3.0 as obsolete.

(3) Utilities in the US have applied MAAP4.0 for SAMG development

Further Clarification:

The MAAP3.0B-ABWR has been used for the Certified ABWR. It is an integrated severe accident analysis code with modular structure that allowed evaluation of severe accidents over a wide range of assumptions and sensitivity analyses. The Certified ABWR severe accident analysis involved both MAAP analysis and special analyses (e.g., debris freezing in the drywell sump shield) to address issues which are beyond the MAAP capabilities. This MAAP/special analyses combination has been reviewed by the USNRC and ACRS and formed the basis for the ABWR certification.

An effective severe accident management (SAM) should be based on plant-specific, risk-informed decisions. For example, a SAM strategy of ex-vessel flooding, which may be effective for a future-generation passive PWR such as AP600, is not considered an effective strategy for the ABWR for the reasons indicated in the response to Track No. 19-009, Item 2. A significant effort by the USNRC SCDAP/RELAP5 code improvement has been related to passive performance issues related to future generation LWRs such as AP600 and SBWR, and to PWR related issues of steam generator tube failure and high pressure melt-ejection. The USNRC has determined that these are high risk-significance issues that require detailed analysis capability to reduce the large uncertainties associated with them so that reliable risk-informed decisions can be made.

The detailed analysis capability of SCDAP/RELAP5 comes at a significant analysis cost. A typical SBO analysis case, for example, takes a week of computer time when analyzed by SCDAP/RELAP5. The USNRC reported in Reference 1 a MELCOR - SCDAP/RELAP code comparison for the Browns Ferry SBO which produced comparable results but took 300 hours of computation time by SDCAP/RELAP as compared to only five hours by MELCOR. This severely limits the potential for using SCDAP/RELAP as a sensitivity analysis tool.

In view of the above differences in analysis capability and cost, the USNRC maintains a two-tier philosophy in severe accident code development: 1) MELCOR as an integrated, flexible, fast-running, and user friendly code that models a wide range of phenomena and their interactions, and 2) detailed codes such as CONTAIN and SCDAP/RELAP5 for detailed modeling of specific areas or phenomena of severe accidents. The above two types of codes produce consistent results, but only MELCOR realistically allows for evaluating uncertainties in phenomena through sensitivity analysis.

Historically, severe accident computer code development has been done in stages. It is expected that the same trend of code improvement will continue both at the USNRC and EPRI. While new code versions provide improvements over older ones, they may be susceptible to coding errors as was indicated in the previous response to this question. Moreover, the development in these codes may be directed to risk-significant issues for a specific type of nuclear power plants. Thus, use of MAAP4.0 by some utilities in the USA as a tool for SAMG analysis does not mean that earlier versions of the code are not adequate for the ABWR SAM analysis. The decision to change the MAAP3.0B-ABWR code which is an already established and flexible code that allowed for analysis of different accident scenario assumptions and phenomenological uncertainties requires careful evaluation of the cost and benefit of such a change. An evaluation of the MAAP3.0B-ABWR and MAAP4.0 capabilities to adequately address Lungmen-specific SAM issues will be made before the FSAR.

Reference:

1. USNRC Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Severe Accidents Subcommittee Meeting, Rockville, Maryland, April 8, 1996.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-010

PSAR Section: 19.7

Question Date: December 3, 1997

PSAR Question:

19.7.1 states that "The SSAR evaluation found that no liner leakage will occur before the containment capability pressure is reached", please provide the evidence.

PSAR Response:

Appendix 19F of the SSAR describes the analysis leading to the above conclusion. The conclusion has been accepted by the USNRC in the FSER, NUREG-1503, Section 19.2.6.2. Appendix 19F of the SSAR analyzes the structural capability of the containment concrete shell and Drywell head, and the leakage potential from the liner plate and penetrations. Please refer to PSAR Section AJ.17 which reproduces the above analysis.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-011

PSAR Sections: Ch 19.4.3.1

Question Date: November 24, 1997

PSAR Question:

1. Please briefly describe the method, using a multiplier to non-mechanistically generate oxidation of the active cladding, for the 100% Metal-Water Reaction simulation. Has the heat energy from the oxidation reaction been considered at the same time?

2. What is the basis for the assumption that the probability of hydrogen combustion can be ignored when the containment is not inerted (e.g., under shutdown or low power operation conditions) ? Any procedures for these conditions to avoid hydrogen combustion ?

3. How long does it take to go from containment un-inerted state to inerted state when necessity arises ? Is it sufficient to prevent hydrogen combustion during the whole time ?

4. Have the same Passive Autocatalytic Recombiners as used in ALWRs been used for Recombiners ?

PSAR Response:

1. The multiplier used for the MAAP-ABWR model increased the number of clad sides, and thus effectively increased the cladding surface area and rate of metal-water reaction. This was not sufficient by itself to cause interaction of 100% of the active fuel cladding, so injection of water to the vessel was started and stopped many times to increase the time available for metal water reaction to occur. It is clear that this type of scenario is unrealistic.

   The MAAP-ABWR code used for the above analysis accounts for the interaction rate, reaction heat generated, and partitioning of the reaction heat between the generated hydrogen, steam, and reactor core components and structures through proper thermal hydraulic analysis. Section 19.4.3.7.1.1 provides the details of the scenario used in the MAAP analysis and the results of this analysis.

2. As indicated in the response to item 3 of Question 19-006, a guideline used to simplify the PRA has been to disregard a group of accident sequences that may lead to early containment failure if they contribute less than 1% of the total CDF. Use of this guideline results in dismissing accidents that may cause containment failure due to hydrogen burning.

   The core damage frequency (CDF) during shutdown is less than 1% of the total CDF (Table A1-4 of the PSAR Appendix A). As indicated in Page A7-17 of the PSAR Appendix A, 97% of the shutdown CDF is due to events occurring during modes 4 (cold shutdown) and 5 (refueling) where the PRA did not take credit for the containment integrity by assuming that the drywell head is removed. Thus, hydrogen burning during these modes does not impact the risk. Consequently, hydrogen burning will impact the shutdown risk only for accident sequences occurring during mode 3 (hot shutdown) which contribute 3% of the shutdown CDF (or less than 0.03% of the total CDF) and only if no inerting of the containment is initiated.

   The containment is not inerted during startup (mode 2; from shutdown to 15% power) following refueling and unplanned shutdowns that require containment entry. The containment may continue to be de-inerted for up to 24 hours after power is increased above 15% (mode 1). In preparation for shutdown, the containment may also be de-inerted for up to 24 hours prior to decreasing power below 15%. The total time in modes 1 and 2 during which the containment is de-inerted is about 3 days or less. Since refueling is done every 18 months, the average refueling mode 1 and mode 2 time during which the containment is de-inerted is no more than 2 days/year. Unplanned shutdowns are projected for Lungmen to be less than 1 shutdown per year. Assuming that containment entry will be needed in 50% of unplanned shutdowns, the average containment de-inerting time due to unplanned outages is no more than 1.5 days/year (0.5 de-inerted shutdowns/year x 3 days/shutdown). Thus the total (refueling and unplanned outages) mode 1 and mode 2 time during which the containment is not inerted is conservatively estimated as 3.5

days/year or less than 1% of the time of plant operation. Should hydrogen release become a real threat or occur during this time, the containment can be re-inerted within 4 hours to prevent hydrogen combustion. The risk of hydrogen burning during shutdown or startup is further reduced by the use of the hydrogen recombiners.

Based on the above considerations, the CDF for accident sequences involving hydrogen burning is expected to be much less than 1% of the total CDF. This is the basis for ignoring hydrogen burning during the time the containment is not inerted.

Please see response to question number 3 for the containment inerting and hydrogen recombiners capability.

3. As stated in Page 19.4-2, the Atmospheric Control System (ACS) is capable of reducing the wetwell and drywell oxygen concentration from atmospheric conditions to less than 3.5% by volume in less than 4 hours. This oxygen concentration limit is maintained during operation. The limit is below the oxygen concentration limit of 5% by volume specified in Regulatory Guide (RG) 1.7 to prevent burning in accidents which result in hydrogen concentrations greater than 6% by volume. The possibility of hydrogen burning is further reduced by use of the hydrogen recombiners. The hydrogen recombiners can be used to maintain the hydrogen concentration below the RG 1.7 limit of 4% by volume which will prevent hydrogen burning before the containment inerting is complete during shutdown or startup.

4. Passive autocatalytic recombiners have been evaluated for use in the Lungmen ABWR but were not selected because of the need for further testing to gain confidence in their reliable performance and approval of their use by the USNRC. The Lungmen ABWR will use thermal recombiners.

No PSAR revision is proposed in response to this question.

ROCAEC Review Comment:

The most important advantage of the Passive Autocatalytic Recombiners is that it does not need the electric power, which contributes significantly to plant safety. Although it needs further testing to gain confidence in its reliable performance, USNRC is considering to use it in ALWR. If USNRC approves its application to ALWR in the near future, will Lungmen ABWR take it into consideration?

Further Clarification:

During the proposal stage for Lungmen 1 & 2, GE proposed the Passive Autocatalytic Recombiner (PAR) as another option to the Thermal Type Hydrogen Recombiner. However, the USNRC has not accepted the PAR at this time due to pending resolution of questions from the ACRS. Further, TPC requires that equipment used in Lungmen must have at least a twelve month demonstration of successful performance in an operating plant. Therefore based upon the above two circumstances, TPC selected the use of the Thermal Type Hydrogen Recombiner for Lungmen NPS Units 1 and 2.

Upon USNRC approval of PAR for ALWR application and satisfactory demonstration of performance in an operating plant, GE would definitely consider using PAR for the control of combustible gases inside the primary containment for future plants. At this point, it would be too late to apply PARs to Lungmen NPS 1 & 2 because of schedule. In addition, the contract for the supply of the thermal hydrogen recombiners for units 1 & 2 has been placed.

ROCAEC Review Comment:

In the PSAR Response to Item 2 of Question 19-011, the following explanation is shown.
"The CDF for accident sequence involving hydrogen burning is expected to be much less than 1% of the total CDF. This is the basis for ignoring hydrogen burning during the time the containment is not inerted."
Furthermore, in the Response to Track Number 19-006 question 3 (ROCAEC further comments), the above guideline (ignore sequences leading to early containment failure if their CDF is < 0.01 * Total CDF) has not been proposed or approved by USNRC. The above guideline can simplify the severe accident analysis a lot.

Question:
Please provide the supporting reasons for application of this guideline.

Further Clarification:

The 1% total CDF guideline is consistent with the USNRC policy statement on the use of PRA for risk-informed regulatory decisions (Reference 1) and its implementation in draft Regulatory Guideline DG-1061 (Reference 2) as discussed below.

PRA provides two primary advantages for safety decisions:

1) A thorough account of plant challenges and their safety consequences.
2) A rational approach for differentiating between risk-significant and non-risk-significant accident sequences.

Because of the extremely large number of possible accident sequences, it has been the general practice to truncate the analysis of non-risk-significant sequences; i.e., those sequences that make a "very small" contribution to the risk (See below for USNRC definition of "very small"). Focusing on analysis of risk-significant sequences enhances better understanding and communication of dominant contributors to the risk. This in-turn leads to more efficient use of resources for risk management. This is a primary motivation for the policy statement of Reference 1. Specifically, in its approval of that policy statement, the USNRC envisioned its implementation to improve the regulatory process in three areas (Reference 2, page 1-1):

- Enhancement of safety decision making by the use of PRA insights.
- More efficient use of agency resources.
- Reduction in unnecessary burdens on licensees.

On implementing the above policy in Draft Regulatory Guide DG-1061, the USNRC indicated that CDF and LERF (large early release frequency) can be used as suitable metrics for making risk-informed regulatory decisions (Reference 2, page 2-3). Section 2.4.2.1, page 2-8, of Reference 2 provides the following CDF and LERF criteria for acceptance of changes to the current licensing basis proposed by a licensee:

- For plants with a mean CDF of less than 1.0E-4 per reactor year, applications will be considered which, when combined with the LERF guidelines described below:
- Result in net decrease in CDF or are CDF-neutral;
- Result in increase in calculated CDF that are "very small" (e.g., CDF increase of less than 1.0E-6 per reactor year); or
- Result in an increase in calculated CDF in the range of 1.0E-6 to 1.0E-5 per reactor year subject to increased NRC technical and management review ....

  AND
- For a plant with a mean LERF of less than 1.0E-6 per reactor year:
- Result in net decrease in LERF or are LERF-neutral;
- Result in increase in calculated LERF that are "very small" (e.g., LERF increase of less than 1.0E-7 per reactor year); or
- Result in an increase in calculated LERF in the range of up to 1.0E-6 per rector year subject to increased NRC technical and management review ....

The above criteria suggest that an increase of 1% of the total CDF and 10% of the LERF are acceptable (because they are "very small") and do not require increased NRC technical or management review.

Lungmen PSAR Appendix A, Table A1-3, page A1-22 show a total CDF of 3.45E-6 per reactor year, and a total LERF of 5.3E-7 per reactor year. Using these estimates, a 1% increase of the total CDF is equal to 3.45E-8 per reactor year. If the accident sequences contributing this 1% increase of the CDF are assumed to lead to early containment failure, and consequently large early release, this will increase the LERF by 3.45E-8 per reactor year, which is equivalent to an increase of 6.5% of the total Lungmen LERF. Therefore, the 1% total CDF truncation criterion used for Lungmen is consistent with the above 1% CDF and 10% LERF acceptance criteria. Please note that the CDF for accident sequences involving hydrogen burning is expected to be much less than 1% of the total CDF. Therefore, disregarding hydrogen burning during the time the containment is not inerted has a very small impact on the CDF and LERF.

References:

1. "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," 60FR42622, USNRC, August 16, 1995.
2.
2. "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Current Licensing Basis - Draft for Comment," Draft Regulatory Guide, Draft DG-1061, USNRC, March 28, 1997.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-012

PSAR Sections: Ch 19.4.3.4

Question Date: November 24, 1997

PSAR Question:

When Firewater Spray is activated, the drywell to wetwell pressure difference is 0.1 MPa and when Firewater Spray is not activated, the drywell to wetwell pressure difference is not more than 0.05 MPa. Why the setpoint of COPS is based on 0.05 MPa and not on the more conservative 0.1 MPa pressure difference ?

PSAR Response:

The choice of 0.05 MPa pressure difference provides longer holdup of fission products within the wetwell airspace than does the choice of 0.1 MPa pressure difference. This allows for more fallout and plateout of the suspended aerosol and for more decay of the fission products relative to earlier release.

As explained in Section 19.4.3.4.1, the setpoint was selected by considering both the desire to delay the release as long as possible, and the desire to minimize the probability of containment structural failure. Three severe accident cases were considered in the selection of COPS setpoint:

1) Cases with no reactor pressure vessel failure which result in wetwell pressurization and have no significant pressure difference between the wetwell and drywell
2) Cases with 0.05 MPa pressure difference which result from sequences with drywell pressurization with no water addition to the containment
3) Cases with 0.1 MPa pressure difference which result from sequences in which the Firewater Spray System is used.

Based on the results of the studies documented in Section 19.4.3.4, the COPS setpoint of 0.72 MPa was selected.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-013

PSAR Sections: Ch 19.4.3.5

Question Date: November 24, 1997

PSAR Question:

Vaughan aerosol plugging model is only applicable to results that the escape path is less than 1 cm (USNRC considers acceptable range). But it has been used for cases longer than 1 cm (~1.6 cm) and shorter duct length (~2 cm) in the PSAR (see page 19.4-51, third paragraph). Please explain.

PSAR Response:

The plugging phenomenon is treated probabilistically in Section 19.4.3.5 in recognition of the uncertainties in the size of the leak and plugging possibility. As pointed out in the review of plugging experiments in page 19.4-51, there is a significant uncertainty in the plugging possibilities for leak sizes larger than 1 cm. For this reason, the probabilistic analysis conservatively assumed no plugging for leak sizes larger than 0.9 cm. Even for leak sizes smaller than 0.9 cm, the probabilistic analysis did not take full credit of plugging by assuming a plugging probability of 0.9 (See 19.4.3.5.2.1.2(3), page 19.4-49, second paragraph). The parametric deterministic analysis reported in Section 19.4.3.5 includes a range of leak sizes beyond the 1 cm size to show the sensitivity of the radioactive release to the leak size.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-014

PSAR Sections: **Ch 19.7 Capacity of Primary Containment**

Question Date: **December 4, 1997**

PSAR Question:

Setting point of Rupture Disk for COPS is 0.72 MPa, which is below the containment capacity. What is the effect during severe accident due to this difference?

PSAR Response:

As discussed in the response to Question 19-012, the setpoint was selected to delay a potential radioactive release while minimizing the potential of containment structural failure. The selection of COPS set point below the containment pressure capacity has three main effects:

1. The potential for containment structural failure is reduced by an order of magnitude
2. The release of volatile fission products is virtually eliminated
3. The timing of the noble gas fission product release is slightly earlier.

The above effects are discussed in Section 19.4.3.4.4, p.19.4-43 and in Section 19.4.3.4.7, p. 19.4-45. Section 19.4.3.4.4 describes several MAAP-ABWR analyses for various accident sequences both with COPS relieving the pressure when the set point is reached, and without COPS pressure relief. The section compares the time of radioactive release and total release for cases with and without COPS. Table 19.4-3, p. 19.4-114 shows significant reduction in the radioactive release for the cases with COPS pressure relief. Section 19.4.3.4.7 presents the significant reduction in containment failure probability as a result of the COPS pressure relief.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-015

PSAR Sections: Ch 19

Question Date: December 26, 1997

PSAR Question:

1. P.19.2-1 listed four key elements and the third one mentioned that further requirements are provided. An example was given in the 2nd paragraph from the last of the same page. However, all the features listed in pages 19.2-2 and 19.2-3 did not identify which ones are used to satisfy the requirements of the 3rd key element. Please explain.
2. The Overview in P.19.2-4 should add a summary description on how the three guarantees and one minimization requirement listed in P.19.1-1 can be met.

PSAR Response:

1. The added selected requirements of the third key element of P.19.2-1 have been derived from consideration of severe accidents that involve reactor vessel melt-through or are accompanied by suppression pool bypass. The following features are presented in pages 19.2-2 through 19.2-4 as examples of design features specifically added to meet these requirements:
   1) Intersystem LOCA prevention provisions (1)(b), page 19.2-2
   2) Large lower drywell floor area to enhance core debris coolability (2)(c), page 19.2-3
   3) ACIWA, for containment cooling and radioactive material removal (2)(d)(ii), page 19.2-4
   4) Lower drywell flooder for debris cooling (2)(d)(iii), page 19.2-4
   5) COPS for controlled pressure relief (2)(e), page 19.2-4.

   Other mitigative features presented in Section 19.2 provide further assurance that the containment function will be preserved for a broad range of accidents. Please note that the examples given in Section 19.2 do not cover all features used to meet the above requirements. Please see response to question 19-016 and Table 19.5-1 for a more complete overview of the features used for severe accident mitigation.

2. The suggested summary is considered an excellent idea to provide a complete picture of the severe accident prevention and mitigation assurance provisions of the Lungmen NPS. Such provisions are not confined to design features only, but extend to operation requirements such as leak testing of the containment, to the design process such as the application of codes and standards to safety systems, ...etc. As indicated in the second paragraph in page 19.2-1, some of these provisions fall under the scope of other PSAR chapters. In fact, the whole PSAR may be considered as an account of how the severe accident prevention and mitigation assurance philosophy of page 19.1.1 is applied in the Lungmen NPS design and operation. For this reason, the PSAR Chapter 19 SRP (Reference 19.1-4) confines the scope of Chapter 19 to only those issues discussed in Sections 19.3 through 19.7.

   The overview of Section 19.2, pages 19.2-1 through 19.2-4, presents a preliminary summary that aims at placing the contents of Chapter 19 in the context of the overall approach to meet the severe accident protection requirements of page 19.1-1. Specifically, the first paragraph of the section defines the four elements used in the PSAR to meet these requirements. This is supplemented by the examples shown under Prevention Features in pages 19.2-2 and 19.2-3 which address items 1 (accident resistance) and 2 (core damage prevention) identified as the two levels for severe accident prevention in P. 19.1-1. Section 19.3 covers accident prevention issues that have been identified in the SRP of PSAR Chapter 19. Section 19.6, Severe Accident Management, also contains actions for accident prevention. The examples shown under Mitigation Features in pages 19.3-3 and 19.3-4 address the third element (challenges to containment, its internal structures, and enclosed equipment). This item is also addressed more fully in Sections 19.4 and 19.6. Sections 19.5 through 19.7 address the fourth element (containment safety margin against severe accidents).

   The above preliminary summary will be updated in the FSAR where it will be supplemented by a table that shows direct relationship between the above requirements, the defense-in-depth provisions for meeting them, and where these provisions are covered in the FSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-016

PSAR Sections: Ch 19

Question Date: December 26, 1997

PSAR Question:

The mitigation discussion in this chapter (Section 19.4) is very confusing and too long (compared with Section 19.5 or 19.6) and some of the levels even go up to nine levels such as 19.4.3.8.2.1.5.2.1 which makes effective review impossible. Please reorganize this section and summarize.

PSAR Response:

Section 19.4 is organized similar to Section 19.2.3 of NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," July 1994. The organization is consistent with the SRP developed for Chapter 19 which used NUREG-1503 as a primary source. To improve the readability and clarity of the section, the summary given below will be considered for inclusion at the beginning of Section 19.4 in the revised PSAR. Restructuring the section by adding similar summaries and moving detailed analyses to appendices, rather than in the main text, will be considered for the FSAR.

Section 19.4 presents evaluations of severe accident mitigation capabilities of the Lungmen NPS. The section covers three primary areas: 1) severe accident challenges to the structural integrity of the containment, containment floor, containment floor drain sumps, RPV pedestal, and equipment inside the containment needed for severe accident mitigation, 2) the Lungmen features that reduce the probability and severity of these challenges, and 3) analysis that was performed to evaluate the effectiveness of these features.

The section contains three main subsections. Section 19.4.1 presents an overview of the containment design with brief description of the containment structures, containment cooling systems, and the Atmospheric Control System (ACS) which provides containment isolation and includes systems for containment inerting, flammability control, containment overpressure protection, and flooding of corium debris that may be released to the lower drywell. Section 19.4.2 presents an overview of in-vessel and ex-vessel potential severe accident scenarios and key phenomena, and highlights the causes of uncertainties in predicting severe accident progression. Section 19.4.3 provides detailed analysis and discussion of eight issues that have been identified in NUREG-1503 and Chapter 19 SRP as key issues to be addressed in evaluating severe accident mitigation capability, namely:

- Hydrogen generation and control (Section 19.4.3.1)
- Core debris coolability (Section 19.4.3.2)
- High pressure core melt ejection (Section 19.4.3.3)
- Containment vent design (Section 19.4.3.4)
- Suppression Pool bypass (Section 19.4.3.5)
- Fuel-coolant interaction (Section 19.4.3.6)
- Equipment survivability (Section 19.4.3.7)
- Protection of containment sumps (Section 19.4.3.8).

The content of each of the above sections is briefly described below.

1) Hydrogen Generation And Control (Section 19.4.3.1, p. 19.4-6)

Section 19.4.3.1 discusses the extent and consequences of hydrogen generation from zirconium-water reaction in the reactor vessel. Mechanistic analysis using the MAAP-ABWR computer code is reported for four scenarios that present different challenges to the containment. The analysis uses conservative metal water reaction modeling assumptions that allow such a reaction to continue in the presence of flow channel blockages and eutectic formation. A non-mechanistic analysis is also presented which assumes reaction of 100% of the active core zirconium cladding. These analyses lead to the following main conclusions.

1) MAAP-code analysis leads to metal-water reaction involving less than 40% of the cladding.
2) Use of the MAAP analysis assumptions indicated above increases the radioactive release if the ac-independent water addition system (ACIWA) is not initiated. However, this increase virtually diminishes

if the ACIWA is initiated. Therefore, the ACIWA provides a protection against uncertainty in the fraction of metal reacting.

3) The non-mechanistic hydrogen release corresponding to reaction of 100% of the zirconium cladding increases the containment pressure to 0.618 MPa which is below the 0.72 MPa setpoint of the containment overpressure protection system (COPS) and the 0.77 MPa level C capability of the containment. Therefore, the COPS elevated setpoint and containment structural capacity provide two more levels of safety assurance against the uncertainty in the fraction of metal reacting.

2) Core Debris Coolability (Section 19.4.3.2, p. 19.4-8)

Section 19.4.3.2 discusses the core debris coolability potential and its impact on core concrete interaction (CCI). Core-concrete interaction could cause vertical erosion of the containment concrete floor, radial erosion of the pedestal walls, and containment pressurization due to the generated heat and released gases. The section provides an overview of the CCI preventive and mitigative features. These include: 1) a large containment floor area to enhance debris coolability, 2) ACIWA and the lower drywell passive flooder which provide both cooling capability and fission product scrubbing, 3) basaltic concrete in the containment floor to reduce the gas release from CCI, 4) COPS to protect against containment overpressurization, 5) large depth of the basaltic concrete floor to protect the containment liner, and 6) thick pedestal walls to ensure its load bearing capability with CCI erosion. The section also presents analyses of the CCI prevention and mitigation capability. Due to the large number of possible CCI scenarios and uncertainties, both probabilistic and sensitivity analyses are performed. Probabilistic analysis uses decomposition event trees (DET) to establish debris coolability scenarios and their probabilities, and pedestal attack scenarios and their probabilities. The DETs identify the key factors that affect these scenarios, e.g., the initial amount of corium released to the lower cavity, the level of corium superheat, if and when cooling water is available for debris quenching, upward heat flux, and whether the debris flows to the suppression pool. Sensitivity to various modeling assumptions is investigated using the MAAP-ABWR code. Finally, the section provides evaluations of the pedestal strength and the effect of using different types of concrete for the pedestal walls. These analyses lead to the following main conclusions:

1) For the core melt sequences that release core material into the containment, the probability of no significant CCI is 0.9.
2) Even for the unlikely cases with significant CCI, axial erosion in 24 hours is less than the sacrificial basaltic bed depth and radial erosion is below the allowable structural limit for the pedestal.
3) For the dominant scenarios with successful operation of the ACIWA to provide water to the core debris, release starts at 24 hours after melt inception for the wide range of assumed upward heat transfer rates. Therefore, the ACIWA provides a level of safety assurance against uncertainty in the upward heat transfer rates.
4) For all sequences with successful operation of the passive flooder, the release time is in the order of 20 hours. Thus, the passive flooder acts as a level of safety against early containment failure given uncertainty in the CCI phenomena.
5) The fission product release for a sequence with CCI is determined primarily by operation of the COPS. The release, which occurs at about 24 hours, is not distinguishable from a case with no CCI. Therefore, the COPS provides protection against uncertainties in the CCI phenomenon.

3) High Pressure Core Melt Ejection (Section 19.4.3.3, p. 19.4-28)

Section 19.4.3.3 discusses the issue of high pressure core melt ejection (HPME). High pressure core melt ejection into the containment could lead to direct containment heating (DCH). The DCH phenomenon involves fragmentation and dispersal of the debris into the containment, with subsequent rapid heat transfer to the containment atmosphere which may cause early containment failure by overpressurization. Direct containment heating has been identified as a PWR severe accident phenomenon which results from HPME. Since BWRs operate at lower pressure and have automatic depressurization systems (ADS), the phenomenon is generally not important for BWRs. However, the significant cooling capability at low pressure in the ABWR also has reduced the importance of low pressure melt ejection and consequently may have increased the relative importance of HPME, thus justifying its evaluation. Section 19.4.3.3 discusses the potential and consequences of DCH in the Lungmen ABWR. The section provides an overview of the DCH preventive and mitigative features which include the ADS, the lower drywell configuration, and the containment structural capacity. Similar to Section 19.4.3.2, both probabilistic and sensitivity analyses are used for evaluating the above features. Probabilistic analysis uses DETs to establish DCH scenarios and their probabilities. Sensitivity of the probability of early containment failure to various modeling assumptions is also investigated. These analyses lead to the following main conclusions:

1) The conditional probability of early containment failure due to DCH given core damage is 1×10^{-3}. This low probability presents the combined effects of the ADS, lower drywell configuration, connecting vents configuration and area, and the containment structural capacity.
2) The conditional probability of early containment failure depends on the containment pressure prior to the HPME. Sensitivity analysis using conservative assumptions that elevate the initial containment pressure show worst-case conditional containment failure probability value of 1.5×10^{-2} given core damage. Thus the containment structural capacity provides protection against uncertainties in the initial containment pressure which may result from uncertainties in the initial steam fraction, hydrogen release, and clearing of the wetwell connecting vents before the DCH event occurs.

4) Containment Vent Design (Section 19.4.3.4, p. 19.4-40)

Section 19.4.3.4 presents the containment vent design which is used to protect the structural integrity of the containment. Containment structural failure by overpressurization leads to uncontrolled radioactive release. Although the containment water spray provides a mitigative feature which reduces the containment pressure and airborne fission products, an overpressure protection system, COPS, is used in the Lungmen ABWR to further scrub the fission products by forcing their passage through the suppression pool before their release, and to terminate the radioactive release when the containment overpressurization threat is removed. Section 19.4.3.4 describes the COPS design, including the basis for selecting the COPS rupture disk setpoint and size. The section also presents MAAP-ABWR analysis that compares the radioactive release timing and magnitude with COPS and without COPS operation. The analysis covers the spectrum of accident scenarios which are used in the PRA (PSAR Appendix A). Finally, the section provides evaluations of the sensitivity of the containment failure probability to COPS setpoint, sensitivity of COPS performance to suppression pool bypass, and the impact of hydrogen burning on the COPS performance. The analysis leads to the following main conclusions:

1) COPS reduces the release of fission products aerosol by several orders of magnitude as a result of the suppression pool scrubbing. As expected, the MAAP analysis shows no reduction (or increase) in the aerosol release for accident sequences involving suppression pool bypass.
2) By limiting the wetwell airspace pressure to the COPS setpoint of 0.72 MPa, the containment structural failure probability is reduced by one to two orders of magnitude.
3) Since the COPS setpoint is lower than the drywell failure pressure of 1.023 MPa, the use of COPS leads to earlier release by up to a few hours depending on the accident sequence. The longer time to release initiation if COPS is not used has a minor impact on the probability of recovery from the accident to prevent the release.

5) **Suppression Pool Bypass (Section 19.4.3.5, p. 19.4-45)**

Section 19.4.3.5 describes the investigation of suppression pool bypass, e.g., due to containment isolation failure, as a containment failure mode. The section discusses the suppression pool bypass prevention features, develops bypass leakage scenarios, and estimates the corresponding consequences. An extensive screening investigation of the suppression pool bypass pathways (containment isolation lines, ex-containment LOCA, wetwell/drywell interface) shows that the wetwell/drywell vacuum breaker leakage presents the only potentially risk significant bypass. Probabilistic analysis using DETs is performed to identify leakage scenarios and probabilities. The DETs account for uncertainties in the vacuum breaker leak size and aerosol plugging of leakage paths. The section also presents MAAP-ABWR analysis of the radioactive release to the environment for five accident scenarios involving vacuum breaker leakage. The above analyses lead to the following main conclusions:

1) Suppression pool bypass does not add significantly to the risk because the bypass areas resulting in increased release are offset by low probabilities of occurrence.
2) The conditional probabilities of no bypass leakage, small leakage (1-10% of volatile fission product inventory), and large leakage (10-20% of volatile fission product inventory) are 0.978, 0.018, and 0.004 respectively.
3) The time of radioactive release initiation does not change with vacuum breaker bypass if the containment spray is initiated. Thus the ACIWA use in the spray mode provides a protection against vacuum breaker bypass uncertainties.

6) Fuel-Coolant Interaction (Section 19.4.3.6, p. 19.4-56)

Section 19.4.3.6 discusses the potential and consequences of molten fuel-coolant interaction (FCI). This interaction refers to the rapid heat transfer from fragmented fuel to water and could cause containment overpressurization, or explosive impulse or water missiles that may challenge the structural integrity of the containment and pedestal. The section covers both in-vessel and ex-vessel FCIs. The section provides an overview of the FCI preventive and mitigative features. These include: 1) in-vessel and ex-vessel structures that result in incoherent relocation of molten corium, 2) ABWR containment configuration which limits the potential for water to be in the lower drywell at the time of vessel failure, 3) drywell connecting vents, and 4) the pedestal and containment structural capabilities. The section presents an evaluation of the probability of drywell flooding prior to vessel failure. The section also reviews energetic FCI tests which involve fuel injection into the coolant and tests for the more benign stratified configuration where the cooling water is injected on top of the fuel as is the case for ABWR severe accidents. Finally, the section presents bounding deterministic calculations of the pedestal loading resulting from explosive steam generation and water missiles, and containment overpressurization. These analyses lead to the following main conclusions:

1) In-vessel FCI presents a negligible risk to the structural integrity of the containment and consequently can be dismissed as a viable containment challenge.
2) The pedestal can stand a peak pressure of at least 0.85 MPa during a steam explosion. It will take more than three times the amount of fuel shown experimentally to participate in an FCI to reach this pressure. Therefore, the pedestal strength provides an adequate margin against uncertainties in energetic FCIs.
3) Water missiles present a negligible risk to the structural integrity of the pedestal or containment and can be dismissed as a viable challenge to these structures.
4) FCI-initiated containment overpressurization does not present a viable containment challenge.

7) Equipment Survivability (Section 19.4.3.7, p. 19.4-73)

Section 19.4.3.7 identifies the equipment important for severe accident mitigation and specifies requirements to ensure their survival during the accident. The section develops a list of equipment needed to terminate accidents in-vessel or ex-vessel from reviews of 10CFR50.34, SECY-90-106, PRA, Emergency Procedures Guidelines, and safe shutdown equipment list. The section presents bounding environmental conditions for the identified equipment based on MAAP-ABWR analysis. These bounding conditions are then compared to the equipment specification to provide a measure of confidence that the necessary equipment would survive the severe accident conditions.

8) Protection of Containment Sumps (Section 19.4.3.8, p. 19.4-85)

Section 19.4.3.8 presents the evaluation of the sump shield which has been designed to protect the drywell sumps and prevent corium ingression through their channels. The Lungmen NPS has two drain sumps in the periphery of the lower drywell floor which could collect core debris during severe accidents if ingression is not prevented.. The section establishes design criteria for the shield material and specifies the shield height, depth below the lower drywell floor, and channel length and height. The section reviews applicable experimental work on flow and freezing of molten debris in narrow channels. The design adequacy to prevent debris ingression is demonstrated using bounding flow and heat transfer calculations.

The evaluations summarized above demonstrate the robustness of the Lungmen preventive and mitigative features against severe accidents challenges and uncertainties. The conservative equipment survivability requirements provided ensure that instrumentation and active as well as passive equipment will be available to terminate and contain such accidents.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-017

PSAR Sections: 19.4.3.1

Question Date: January 20, 1998

PSAR Question:

1. Does Lungmen plant design use igniters to maintain oxygen concentration?
2. Are the recombiners capable of avoiding local hydrogen burn?

Response:

1. The Lungmen NPS does not use igniters to maintain oxygen concentration. The containment is inerted by purging with nitrogen.
2. Recombiners reduce the chance of local hydrogen burning in the Lungmen NPS by reducing the hydrogen and oxygen content of the containment atmosphere. However, local hydrogen burn is prevented primarily by maintaining an inerted containment with its low oxygen concentration well mixed in the containment atmosphere.
   As stated in PSAR Section 6.2.5.2.1, in an oxygen-deficient, well-mixed containment atmosphere, mixing of any hydrogen generated by a design basis LOCA is not required to prevent local hydrogen burn. Any oxygen evolution from radiolysis following a design basis LOCA is very slow such that natural convection and molecular diffusion are sufficient to provide uniform mixing of the oxygen in the drywell and suppression chamber atmosphere. The mixing will be further promoted by operation of the containment sprays and the drywell cooling fans.
   Furthermore, the ACS maintains an oxygen deficient atmosphere (<3.5% by volume). This provides a safety margin against measurement errors and inadequate mixing of the containmnet atmosphere. The above provisions prevent local and global hydrogen burning from occurring. The hydrogen burning risk is further reduced by activating the FCS recombiners to process the combustible gases drawn from the primary containment.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-018

PSAR Sections: 19.4.3.2

Question Date: January 20, 1998

PSAR Question:

In section 19.4.3.2.2.7 Impact of Pedestal Concrete Selection (p. 19.4-27, line 16), it is mentioned that the type of concrete to be used in the pedestal is not specified.

a) Is the type of concrete for the pedestal determined now?

b) Please provide the comparison results of various types of concrete used in the pedestal.

Response:

a) The pedestal concrete type has been defined. It is normal weight structural concrete with compressive strength of 27.6 MPa at 90 days. Please see section 3.8.3.6.2, page 3.8-27 of the PSAR for further information on the pedestal concrete.
The above concrete is a limestone sand concrete. This type of concrete was evaluated as a candidate for use in the pedestal wall during the Standard ABWR certification. The results of this evaluation are presented in Section 19.4.3.2.2.7, page 19.4-27. As concluded in that section, use of limestone sand concrete for the pedestal sidewall will reduce the rate of sideward ablation by corium-concrete interaction (CCI) relative to that presented in Section 19.4.3.2 analysis. The rate of non-condensable gas generation may be slightly higher for limestone sand concrete than the basaltic concrete used in the drywell floor. However, as concluded in page 19.4-28, because of the relative CCI areas of the sidewall and the drywell floor, the impact of differences in the rate of non-condensable gas generation will be small, and the conclusions of the uncertainty analysis of Section 19.4.3.2.2.8 will not be affected. Based on these conclusions, no new analysis for the selected concrete type will be performed.

b) The comparison between the two types of concrete considered for the pedestal is reported in Section 19.4.3.2.2.7. As indicated in that section, the basaltic concrete used for the sacrificial bed on the drywell floor has the advantage of reduced release of non-condensable gases, while the limestone common sand concrete has the advantage of slower erosion by the corium. As seen from the discussion in the third paragraph of that section, these differences have a second order effect on the pedestal performance due to the small area of the pedestal side wall relative to that of the drywell floor.

PSAR Chapter 19 will be revised to reflect the response to part (a) of this question

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-019

PSAR Sections: 19.4.3.7

Question Date: January 20, 1998

PSAR Question:

1. In page 19.4-84 (13) Temperature Instrumentation, "The GE standard practice is to use thermocouples rated to 575 K and 14 MPa. These ratings are well above the drywell and wetwell thermodynamic loads experience."
   a) In Fig. 19.4-28d (Drywell Temperature for Ex-Vessel Low Pressure Core Melt), the drywell temperature exceeds 575 K after 65 hours. Please discuss the equipment survivability and reliability.
   b) No discussion was made on the in-vessel thermocouple survivability. Please add such discussion in the text.

2. In line 6 of page 19.4-85 "Both wetwell and drywell radiation sensors are qualified to 595 K and 0.65 MPa. therefore, there will be no threat to the performance of the wetwell radiation sensor."
   a) In Fig. 19.4-26c, 19.4-27c, and 19.4-28c, all the wetwell pressures reach 0.72 MPa and actuates COPS. Please discuss the survivability and reliability of the wetwell radiation sensors.

3. In line 11 of page 19.4-85 "The COPS limits the drywell pressure to 0.72 MPa. This is only slightly over the qualification pressure and should not damage the sensors."
   a) The COPS limits the wetwell pressure to 0.72 MPa, instead of drywell pressure. In Fig 19.4-28a, the drywell pressure reaches 0.83 MPa which is well above the equipment qualification pressure (0.65 MPa). Please discuss the survivability and reliability of the drywell radiation sensor.
   b) Please provide a table with comparison of the qualification condition and the expected severe accident conditions.
   c) Please describe the improvement of instrumentation in the ABWR design concerning severe accident.

Response:

1.a) PSAR Figure 19.4-28d is the same as SSAR Figure 19E.2-29d. A review of SSAR Fig. 19.E.2-29d and the MAAP-ABWR results for the low-pressure melt accident sequences used to establish the figure revealed that the figure is incorrect. Specifically, none of these accident sequences leads to temperature above 560 K. Therefore, the drywell temperature is not expected to exceed the drywell thermocouples qualification temperature of 575 K.
Figure 19.4-28d will be revised to represent the correct drywell temperature envelope.

b) The reason for not discussing survivability of in-vessel thermocouples is that there is no in-vessel thermocouples in the ABWR. The last paragraph of PSAR Chapter 7, page 7.5-11, discusses the regulatory basis for not requiring core temperature thermocouples in BWRs. As stated in the first line of page 7.5-12, "Instrumentation other than RPV water level indication is not required to assure indication of adequate core cooling."

2.a) As explained in the second paragraph of page 19.4-85, the wetwell radiation sensors are located in shafts embedded in the primary containment wall and are isolated from the primary containment environment by a substantial amount of concrete. This provides a high degree of confidence that the wetwell radiation sensors will survive the bounding severe accident conditions and will provide reliable radiation levels over the duration of these conditions.

3.a) We concur with the statement that COPS limits the wetwell (not the drywell) pressure to 0.72 MPa. The statement will be corrected in the revised PSAR.

Figure 19.4-28a presents the envelope of drywell pressure for the accidents analyzed. The pressure peaks shown in the figure represent different accidents. As seen in Fig 19.4-28a, the drywell pressure exceeds 0.65 MPa over a period of less than 3 hours for any of the accidents enveloped by the figure. The drywell temperature during this time is less than 520 K which is 75 K below the qualification temperature. It is also noted that the pressure of 0.83 MPa is less than 30% above the qualification value of 0.65 MPa.

In the FSER of the certified ABWR design, the USNRC indicated that short time of exposure to beyond-design-basis environment is an acceptable criterion to demonstrate equipment survivability under severe accidents. This is based on the universally recognized fact that design basis qualification and analysis are conservative and more realistic assessment in similar situations shows a substantial margin. For example, the SSAR ABWR containment pressure has to increase to ~ 2- times its design pressure before the containment Service Level C stress is reached.

Based on the above criterion, the drywell relatively benign environment conditions for the short duration indicated provide a reasonable level of confidence that the drywell radiation sensors will survive severe accident conditions and will provide reliable radiation levels over the duration of these conditions.

It should be noted that failure of the drywell radiation sensors has an insignificant impact on public consequences from severe accidents, since they are backed up by the protected wetwell radiation sensors. The wetwell sensors will survive the severe accident conditions as discussed in part (a) of this question. Since the radioactive release path is through the suppression pool then the COPS stacks, the wetwell radiation sensors will provide a more direct indication of the released radiation.

b) Table 1 provides the locations, qualification temperature and pressure, and maximum severe accidents temperature and pressure for the sensors of Table 19.4-7.

c) The ABWR has the following main improvements in instrumentation used for severe accidents prevention or mitigation:

i) Use of four independent divisions of self-tested Safety System Logic and Control (SSLC) instrumentation designed on the basis of two out of four actuation logic. Combined with quarterly testing of the Essential Multiplexing System and

SSLC to discover faults that are not identified by the continuous self-test, the SSLC reliability has been significantly improved.

ii) Use of two divisions for remote shutdown operation as opposed to only one division used by all BWRs except one (Limerick).

Table 1 - Design Basis Qualification / Maximum Severe Accident Pressure and Temperature Comparison (1)

Sensor	Location (2)	Qualification	Maximum Severe Acc.
		Temp (K) / Press (MPa)	Temp (K) / Press (MPa)
RPV WL (3)	Outside containment		
RPV Press	Outside containment		
SP Temp.	SP	575 / 14	430 / 0.72
WW Rad.	WW - Embedded in containment wall	595 / 0.65	430 / 0.72 (4)
DW Rad.	DW	595 / 0.65	560 / 0.83 (5)
DW/WW H2	Outside containment		
DW/WW O2	Outside containment		
DW Temp.	DW	575 / 14	560 / 0.83
DW Press.	Outside containment		
WW Press.	Outside containment		
DW WL.	Outside containment		
WW WL	Outside containment		

(1) Table contains sensors identified in Table 19.4-7.
(2) Sensors located outside the primary containment are not exposed to severe accident environment and consequently are expected to survive the severe accident environment.
(3) WL = water level sensor
SP = suppression pool
DW = drywell
WW = wetwell
Rad. = radiation sensor
(4) Sensor is isolated from the ww environment by a substantial amount of concrete and consequently will survive the severe accident pressure.
(5) Pressure exceeds 0.65 MPa for less than 3 hours and at temperatures less or equal to 500 K.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-020

PSAR Sections: 19.0

Question Date: March 2, 1998

PSAR Question:

In Chapter 19.0, a number of references are made to frequency without an expression of confidence level. Where confidence levels are not specifically identified, what confidence level should be assumed?

Response:

The frequency and conditional probability estimates in Chapter 19 are quoted from Appendix A of the PSAR, PRA, where only point estimates have been calculated. Uncertainty analysis and confidence interval estimation are planned for the FSAR. Since the uncertainty analysis involves complex sums and products of random variables that have different statistical distributions, an evaluation of the confidence levels can not be provided at this time.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-021

PSAR Sections: 19.1

Question Date: March 2, 1998

PSAR Question:

Reference is made to Chapter 19 of the Standard Review Plan (SRP) in defining the specific accidents and accident phenomena that should be evaluated. However, Chapter 19 of the SRP issued by the U.S. Nuclear Regulatory Commission (USNRC), Revision L dated March 27, 1997, deals only with licensing amendment requests submitted by existing plants for which probabilistic risk assessment (PRA) is used in risk-informed decision making. Is the Lungmen SRP Chapter 19 different from that issued by the USNRC? If so, please provide the Lungmen SRP chapter 19. What additional regulations, regulatory guidance, or other criteria have been employed in defining the scope of Chapter 19? What additional information relevant to severe accident analysis will be contained in the Lungmen FSAR which is not contained in the PSAR?

PSAR Response:

The Lungmen design is committed to comply with only those USNRC applicable regulations issued before June 24, 1996. As can be seen from the issuance date and purpose of the USNRC Chapter 19 of the SRP noted in the question, the USNRC Chapter 19 of the SRP is not applicable to Lungmen. An SRP addressing severe accident anlysis, Chapter 19, specifically for Lungmen NPS has been prepared and is attached.

The attached SRP is based on the USNRC Final Safety Evaluation Report (FSER) for the certified Standard ABWR. Although the attached SRP is different from that issued by the USNRC in March 27, 1997, the FSER used the Standard ABWR PRA for drawing risk-informed conclusions. The Lungmen SRP Chapter 19 identifies the key severe accident issues discussed in the FSER.

The attached SRP has been the primary source for defining the scope of PSAR Chapter 19. Unless explicitly stated in the Lungmen PSAR Chapter 19, no additional regulations, regulatory guidance, or other criteria have been employed in defining the scope of Chapter 19. Please see response to Question 19-016 for the basis of PSAR Chapter 19 contents and organization.

Please note that the Lungmen PSAR also differs from the certified ABWR Standard Safety Analysis Report (SSAR) in that the Lungmen PSAR contains the PRA in Appendix A and the resolution of Unresolved and Generic Safety Issues and TMI-related Issues in Appendices 1A, 1B, and 1C of Chapter 1. In contrast, the SSAR contains the PRA and resolution of the above issues in Chapter 19.

As stated in Section 19.7 of the PSAR, the FSAR will contain additional analysis related to the containment capacity evaluation. Please see Section 19.7.2 for the specific analyses planned. The FSAR will also provide more specific information on severe accident management (Section 19.6) and will update the severe accident assessments of Sections 19.3 and 19.4 as necessary to reflect new PRA assessments or insights that will be reported in Appendix A of the FSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-022

PSAR Sections: 19.2

Question Date: March 2, 1998

PSAR Question:

1. Please provide information to explain how the defense-in-depth philosophy as well as independence, redundancy, and diversity requirements are implemented in the design of the Lungmen NPS for the vessel pressure control function in preventing the occurrence of severe accidents.

2. On page 19.2.3, it is indicated that only one RBCW/RBSW ESF division is capable of removing heat loads associated with operation of ECCS pumps and only one division is capable of removing suppression pool heat during LOCA. Please explain how the independence and redundancy requirements are satisfied for the heat removal function provided by the RBCW/RBSW.

3. Please identify the pressure head capacity of the ACIWA.

PSAR Response:

1. Reactor vessel pressure control in Lungmen NPS is accomplished through three paths:

   i) Turbine bypass valves (TBPVs), where steam is discharged from the main steam lines (MSL) directly to the main turbine condenser.
   ii) RCIC, where the steam is discharged through the RCIC's turbine to the suppression pool (SP).
   iii) Safety/Relief Valves (SRVs), where the steam is discharged through piping to the SP.

   There are10 TBPVs and 18 SRVs in each of the Lungmen ABWR units. The SRVs are distributed between the four MSLs (4, 5, 5, and 4 SRVs in

MSL A, B, C, and D respectively). Eight of the SRVs are assigned to the Automatic Depressurization System (ADS). Each MSL contains 2 ADS valves which are separated within a MSL by a non-ADS valve.

The SRVs provide three pressure control modes:

i) Overpressure relief operation, where the valves are opened using pneumatic actuators upon receipt of automatic or manually-initiated signal. The pressure set points for the automatic relief mode of operation are lower than the safety operation set points (See next item).
ii) Overpressure safety operation, where the valves function as safety valves to limit the RPV pressure to less than the ASME code limit for the RPV. In this mode, the valves are self actuated when inlet steam pressure is high enough to overcome the retaining spring and frictional forces. The valves are divided into five pressure set point groups.
iii) Automatic depressurization by the ADS; where the ADS valves open automatically as part of the ECCS to allow LPFL operation.

Each SRV is provided with its own pneumatic accumulator and inlet check valve. The accumulator capacity is sufficient to provide one SRV actuation at normal drywell pressure. Each ADS SRV is equipped with another pneumatic accumulator and check valve. The accumulator capacity is sufficient for one actuation at drywell design pressure or five actuations at normal drywell pressure, whichever is more demanding. Makeup supply for the accumulators is provided by the safety-related portions of the nitrogen gas supply system which contains two redundant safety-related nitrogen gas supply trains. Each train has 10 high-pressure nitrogen gas bottles. The capacity of the nitrogen bottles in each train is about 16 times that is needed to open the 8 ADS valves. After the ADS valves are open, the other train can provide enough nitrogen to substitute for 7-day leakage from the valve accumulators.

The SRVs can be operated in the relief mode by remote-manual control from the main control room. Four SRVs can also be operated from the remote shutdown panel.

The number of SRVs and TBPVs required to depressurize the reactor given an accident initiating event depends on the event and whether the MSIVs are closed (isolated events) or open (non-isolation events) as shown in the following summary:

- Small and medium LOCA: 3 SRVs
- Transients with scram: 3 SRVs with LPFL or condensate cooling, 8 SRVs for ACIWA cooling
- Pressure relief: Isolation Events 6 SRVs, Non-isolation: 6 SRVs, < 6 SRVs with one or more open TBPVs.
- ATWS: Isolation Events: 15SRVs, Non-isolation events: 15 SRVs, < 15 SRVs with one or more open TBPVs.

As seen above, the Lungmen ABWR pressure control function has significant defense-in-depth, diversity, and redundancy that ensures its high reliability.

2. The cited statement will be corrected to read as follows (corrections underlined):

"Three 100% RBCW/RBSW ESF division<u>s</u>, with <u>any</u> one division capable of removing <u>all</u> heat loads associated with operation of ECCS pumps and <u>any</u> one division capable of removing suppression pool heat during LOCA."

The above divisions are electrically, mechanically, and hydraulically independent. The structures housing each RBCW and RBSW division have three-hour fire rated boundaries. In addition, each division is protected from flooding, spraying, steam impingement, pipe whip, jet forces, fire from other divisions, and the effect of any non-seismic Category I equipment failure. The RBCW and RBSW systems are designed to meet the foregoing design basis with or without preferred AC power available and with a single active failure.

3. The ACIWA pump pressure capacity is 0.862 MPa (Flow rate: 674.4 m^3 /hr).

Track Number: 19-023

PSAR Sections: 19.3

Question Date: March 2, 1998

PSAR Question:

As part of the U.S. Nuclear Regulatory Commission (USNRC) certification requirements for the Evolutionary Light Water Reactor (LWR) designs, unresolved and generic safety issues must be addressed and compliance with technically relevant portions of the Three Mile Island requirements set forth in 10 CFR Part 50.34(f) must be demonstrated. Resolution of these issues for the standard Advanced Boiling Water Reactor (ABWR) was included in the Standard Safety Analysis Report (SSAR) and reviewed by the USNRC. How are these issues being resolved for the Lungmen NPS? Will the resolution for any of the technical issues applicable to the severe accident prevention be affected due to the differences in design between the ABWR and the Lungmen NPS? If so, please provide information explaining the differences in resolution of the affected issues.

PSAR Response:

As stated in the response to Question 19-021, unresolved safety issues and TMI related issues are discussed in Appendices 1A, 1B, and 1C of Chapter 1 of the PSAR. Please refer to these appendices for resolution of these issues in the Lungmen NPS.

The above issues are resolved for Lungmen using the standard ABWR approach except for one exception: Unresolved Safety Issue A-44, "Station Blackout." As seen in the PSAR Section 1C.2.16, page 1C-34, the Lungmen NPS uses an independent Safety-Grade swing EDG as an alternate AC electric power source instead of the Non-Safety Grade combustion turbine generator (CTG) used in the standard ABWR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-024

PSAR Sections: 19.3.1

Question Date: March 2, 1998

PSAR Question:

1. Shutdown decay heat removal was one of the Unresolved Safety Issues (A-45). This issue is resolved for the existing U.S. plants as part of the Individual Plant Examination (IPE) program. Please provide specific information to demonstrate resolution of this issue for the Lungmen NPS.

2. The Lungmen NPS is designed to minimize challenges by accident initiating events. In addition, redundant and diverse systems have been included in the Lungmen NPS design to provide various safety function s(e.g. reactivity control, vessel pressure control, vessel water level/Inventory control, decay heat removal, etc.) to mitigate the effects of accident initiating events. The adequacy of the design with respect to these protection features can also be measured by the following probabilistic parameters:

   - Frequencies of accident initiators
     Conditional likelihood of safety function failure given occurrence of an accident initiator. This can be further divided into:
     Conditional failure likelihood of the scram function given occurrence of
     an accident initiator.
     Conditional failure likelihood of the pressure relief function given occurrence of an accident initiator (with success and failure of the scram function separately).
     - Conditional likelihood of a transient-induced LOCA given occurrence of an accident initiator.
     - Conditional failure likelihood of the high pressure injection function given occurrence of an accident initiator and successful scram function (with and without transient-induced LOCA).
     - Conditional failure likelihood of the vessel

depressurization function given occurrence of an accident initiator, successful scram function, and failure of the high pressure injection function (with and without transient-induced LOCA).

- Conditional failure likelihood of the low pressure injection function given occurrence of an accident initiator, successful scram function, failure of high pressure injection, and successful vessel depressurization.
- Conditional failure likelihood of the decay heat removal function given occurrence of an accident initiator and successful scram function (with vessel water inventory being maintained separately by high pressure injection and low pressure injection).
- Conditional likelihood of core damage given occurrence of an accident initiator.
- Conditional likelihood of core damage given a transient-induced LOCA.

Based on the results of the analysis performed for the Lungmen NPS, please provide the estimated values of the preceding parameters by accident initiator.

3. This section addresses only four evolutionary LWR certification issues related to severe accident prevention. Please provide rationale for not addressing such Lungmen-specific hazards as earthquake and typhoon in this section. How is the defense-in-depth approach incorporated in the design of the Lungmen NPS for the prevention of severe accidents resulting from these hazards?

Response:

1. According to NUREG-0933, 12/31/88, Section 2, "Task Action Plan Items," page 2.A.85-2 paragraph 2, one of the alternatives proposed by the USNRC staff to resolve USI A-45 was to have each licensee perform a risk assessment for its plant. This assessment would be done in conjunction with the IPE program. Available options for acceptable risk assessments include a Level-1 PRA.

   Appendix A of the PSAR contains a Level-3 PRA for the Lungmen NPS in response to the USNRC Policy Statement on Severe Accidents. The scope of the Lungmen PRA covers a broader spectrum of challenges and consequences than an IPE. Included within the Lungmen PRA is consideration of loss of shutdown heat removal. The Lungmen PRA demonstrates that the Lungmen risk goals are met and identifies risk significant systems, structures, and components to ensure that their reliability is maintained via the Integrated Reliability Assurance Program (IRAP). Therefore, USI A-45 is considered resolved for the Lungmen NPS.

It is to be noted that NUREG-0933, Appendix B, Revision 15, 12/31/92, does not include Issue A-45 because it is considered resolved with no new regulatory requirements by the USNRC. For this reason, the USNRC did not require the issue to be addressed in the SSAR. However, the cited NUREG-0933 defines a related generic issue, GI A-31, "RHR Shutdown Requirements." Please see PSAR Chapter 1, Appendix 1C, Section 1.C.2.10, page 1C-24, for resolution of GI A-31 for the Lungmen NPS.

2. We concur with the statement that probabilistic parameters like the ones cited in the question provide a quantitative insight into the adequacy of the Lungmen safety approach. Appendix A of the PSAR contains the Lungmen PRA which covers internal and external initiating events and evaluates the risk at power as well as shutdown. Please refer to the Executive Summary of Appendix A for an overview of the PRA scope, important accident sequences, and risk important equipment and human actions. The Executive Summary reports the contribution of various initiating events to the core damage frequency and identifies which of the failure sequences specified in the question are important. Please refer to Attachment AB of Appendix A for a more detailed account of the internal initiating events (Table AB.3-1) and for accident sequence event trees. System fault trees are contained in Attachment AA of Appendix A.

3. External events PRA, including earthquake and typhoon, are addressed in Appendix A, Attachments AC through AF. They are outside the scope of Chapter 19 as defined in the Lungmen Chapter 19 SRP. Please see response to Questions 19-016 and 19-021 for further information on the contents of Chapter 19.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-025

PSAR Sections: 19.3.1.1

Question Date: March 2, 1998

PSAR Question:

1. Redundant and diverse actuation signals are provided in the Lungmen NPS design for the scram function. However, all scram signals will not be generated or applicable following all accident initiating events. Please provide a list of the diverse scram signals, by accident initiators, which are designed to be available during plant response to these events.

2. In the event of failure of high pressure coolant injection following failure of control rod insertion and successful SLCS injection, vessel depressurization and low pressure coolant injection are necessary to mitigate the consequence of this event scenario. As the Low Pressure Core Flooder System starts to inject after the reactor pressure vessel (RPV) is manually depressurized, the continued addition of a large quantity of unborated water would tend to displace a portion of the borated water previously in the reactor core region since they may not initially mix well. Large power oscillation may result from this "displacement" of borated water by the cold, unborated water. How will the Lungmen NPS be designed and operated to prevent or reduce this type of power oscillation causing the fuel failure limit to be exceeded?

3. What are the estimated ATWS frequencies and ATWS core damage frequencies for the various external events?

PSAR Response:

1. The list of diverse scram signals, by accident initiators, which are available during plant response to these events are shown in the attached Table 1.

2. SLC injection at 22.7 m3 /hr (100 gpm) provides backup reactor shutdown capability independent of the normal reactivity control system as required by 10CFR Part 50, Appendix A, GDC 26. SLC is sufficient to bring the reactor from full power to a cold subcritical condition without control rod movement, at any time in a core cycle, and at design basis conditions with the reactor in the most reactive xenon-free state and maintain the reactor shutdown. The SLC injects boron which produces a minimum concentration of 850 parts per million (ppm) by weight of natural boron in the reactor at 20°C. To allow for potential leakage and imperfect mixing in the reactor, an additional 25% (220 ppm) is added to the above requirement, resulting in a total requirement of greater than or equal to 1070 ppm. By design, the SLC system injects borated water into the HPCF Loop B discharge piping downstream of the HPCF pump inside the containment testable isolation check valve. This arrangement provides desirable boration of the cold HPCF water that comes in at the top of the reactor core.

For an ATWS event, a requirement to shutdown the reactor with the reactor not shutdown causes entry into the plant's Emergency Operating Procedures (EOPs). EOPs provide instructions to insert control rods, control reactor pressure and water level, and inject boron if necessary to shutdown the reactor. For an ATWS with HPCF failure following failure of control rod insertion (and Alternate Rod Insertion), as well as all other attempts to insert enough control rods within the three minute time period, the Lungmen ATWS logic will continue to sense that the reactor power is not low (Startup Range Neutron Monitor sensed power level ≥ 6% power and High Reactor Pressure for three minutes) and will automatically inject boron with SLC initiation.

With a failure of all high pressure makeup, EOPs would require that the reactor be depressurized so that low pressure systems may be used to control reactor water level. EOPs based on the BWR Owners' Group Emergency Procedure Guidelines (EPGs) specifically address the potential for cold water related power excursions for scram failure events by limiting the rate of cold water injection (e.g., LPCF injection) during and following reactor depressurization. The purpose of limiting the rate of cold water injection is to take advantage of the negative reactivity feedback from the increased void fraction in the core region.

Assessments of these conditions to support EPG development concluded that a short-term power surge may occur, but it would not develop into a reactor instability (oscillation) even if borated water was not injected by SLC into the core. Large power oscillations may develop at low flow and high power conditions, but with sufficient core void fraction or boron in the reactor there is not sufficient power to produce a large power oscillation due to controlled cold water injection.

It should be recognized that the probability of the accident postulated in the question is negligibly small, since it involves the failure of multiple redundant and diverse safety systems for reactor shutdown and cooling. The Lungmen NPS design includes ATWS mitigation features of Alternate Rod Insertion, automatic Reactor Internal Pump (RIP) Trip, higher flow rate and boron-concentration SLC, and automatic initiation of the Feedwater Control System (FWC) runback, (which reduces water level and results in the rapid reduction of core inlet subcooling). The Lungmen NPS also has upgraded high pressure ECCS (2 HPCF and safety grade RCIC). These Lungmen design features and operation with EOPs based on the BWROG EPGs are effective in limiting the magnitude of possible reactor power oscillations for ATWS events.

3. Appendix A of the PSAR shows that of the four analyzed external events (seismic, fire, internal flooding, and typhoon) only ATWS initiated by seismic events has a non-negligible contribution to the CDF of the external event. The CDF due to seismic-induced ATWS is 4.5E-7 / reactor year (Total seismic CDF is 3.14E-6 / reactor year). The primary cause of the seismic ATWS is structural deformation that prevents control rod insertion in the core. The ATWS CDF due to the other three external events is negligible. This is to be expected in view of the high degree of redundancy, diversity, and separation of the reactor shutdown systems and the instrumentation and logic channels.
The PRA in Appendix A is quantified using minimum cutsets that lead to core damage. The annual frequency of ATWS events, whether they lead to core damage or not, is not readily available from these cutsets. Separate calculations will be performed to provide these frequencies in the updated PSAR.

TABLE 1 PSAR Question 19-025

Initiating Events	Scram Sensors For Initiating vents									
	APRM	Core Flow Rapid Coastdown	Reactor Pres. Hi	Reactor Low Level	Drywell Hi Pres.	MSIV Closure	Supp. Pool Hi Temp.	TSV Closure	TCV Fast Closure	Remarks
Loss of FW Heating	X									1
Pressure Regulator Fails High				X		B		O		
IORV							X			
Feedwater Controller Failure-Maximum Demand				X				O		
Pressure Regulator Fails Low	X		B							
Turbine Bypass/Control Fails closed			X	B						18
Load Rejection w/ Bypass									O	2
Load Rejection w/o Bypass	B								X	
Turbine Trip w/ Bypass									O	2
Turbine Trip w/o Bypass	B							X		
MSIV Closure (partial one)	X		X							3
Loss of Condenser Vacuum			B			X		O		4
Loss of Non-Emergency AC Power (grid or Aux)				X					O	5
Loss of all FW Flow				X						
FW Line Break				X	B					
Inadvertent Rod Insert										6
RPS Fault										17
Manual or Auto scram due to misc. plant occurrences										17
MSIV closure (all)	B		B			X				
Loss of Feedwater				X						
Recirculation Pump Trip (one, multiple, all)		X						O		7
Recirculation Flow Control Failure-Increasing Flow	X									8
Recirculation Pump Failure										7
Rod Withdrawal Error at Low Power										9
Rod Withdrawal Error at Power										10
Inadvertent Startup of Idle Recirc. Pump.										16

X: Primary signal to initiate the reactor scram.
B: Backup signal to initiate the reactor scram.
O: Scram occurs on this signal if bypass valves fail to open during the events.

TABLE 1 PSAR Question 19-025 (Continued)

Initiating Events	Scram Sensors For Initiating Events									
	APRM	Core Flow Rapid Coastdown	Reactor Pres. Hi	Reactor Low Level	Drywell Hi Pres.	MSIV Closure	Supp. Pool Hi Temp.	TSV Closure	TCV Fast Closure	Remarks
Control Rod Drop										11
Control Rod Ejection										12
Inadvertent HPCF Startup										13
Feedwater Flow In Core										14
Small Line Break Inside or Outside Containment										15
Steamline Break Outside Containment				B		X				
RWCU Break Outside Containment				X						
Interfacing System LOCA				X						
DBA LOCA (spectrum)				X	B					
RPV Rupture				X	B					

X: Primary signal to initiate the reactor scram.

B: Backup signal to initiate the reactor scram.

O: Scram occurs on this signal if bypass valves fail to open during the events.

Table 1 PSAR Question 19-025 Remarks:

1. Because this event is very slow, the operator action or automatic SCRRI will terminate this event. A loss of 55.6° C feedwater temperature is analyzed to bound this event.
2. Automatic opening of the turbine bypass valves during load rejection or turbine trip will inhibit automatic reactor scram and RPT.
3. A closure of a single MSIV at any given time will not initiate a reactor scram directly. Credit is taken for the operation of reactor high pressure or flux signals to initiate a reactor scram if reactor power is greater than 80% power.
4. PSAR section 15.2.5.3.1. is based on Standard ABWR design, reactor scram is not expected to occur on turbine trip with bypass on condenser low vacuum.
5. Reactor is expected to scram on low reactor level for Lungmen due to loss of Condensate Pumps, Lungmen PSAR Table 15.2-16 analysis event is based on Standard ABWR design.
6. This event describes about the control rod removal error during refueling. The event considers the possibility of inadvertent criticality due to complete withdrawal or removal of the most reactive rod during refueling. The refueling interlock will assure the core shutdown margin is maintained.
7. Reactor scram will be initiated after tripping of all RIPs. Tripping of one or three RIPs, or one RIP seizure event will not cause the reactor to scram, core power will be settled at its steady state after the event.
8. This event assumes fast runout of all RIPs. Reactor power will be settled at new steady state if one RIP experienced a fast runout.
9. The startup range neutron monitor (SRNM) has a period-based trip function that stops continuous rod withdrawal by initiating a rod block. The period-based trip function also initiates a scram if the flux excursion generates a period shorter than 10 seconds.
10. The Automatic Thermal Limit Monitoring (ATLM) operating thermal limit protection function of either MCPR or MAPLHGR protection algorithm stops the rod withdrawal or core flow increase when either operating limit is reached. There is no operating limit violation due to this preventive function.
11. The performance of the FMCRD separation-detection devices and the rod block interlocks virtually preclude the cause of a rod drop

accident.

12. The FMCRD brake mechanism prevents the rod ejection event from occurring.
13. This event is a mild transient. The flux level settles out slightly below operating level during this event.
14. These events are presented in decrease in reactor coolant temperature and increase in reactor pressure analysis.
15. This event is bounded by the postulated events. Termination of the analyzed event is dependent on operator action. The action is initiated with the discovery of the unisolable leak. The action consists of the orderly shutdown and depressurization of the reactor vessel.
16. This transient assumes the interlock fails to prevent restart of the RIP. The overcurrent protection logic trips the electrical bus. One or two more RIPs are tripped due to the bus trip. Reactor power settles at the steady state after the RIPs have been tripped.
17. These events are not specifically described in Chapter 15 of PSAR.
18. This event assumes that one of the control valves fast closed at power.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-026

PSAR Sections: 19.3.1.2

Question Date: March 2, 1998

PSAR Question:

1. During the initial operation of RCIC following an SBO event, the operators are instructed by EPG to manually control the reactor pressure by opening and closing the safety/relief valves (SRV). How many SRV operations (i.e., opening/closing cycles) are necessary until the rate of steam generation is within the range of steam demand by the RCIC turbine? How do the capacities of the SRV accumulators and N_2 backup bottles compare to the total N_2 requirement to support the initial SRV pressure control operations during RCIC injection, vessel depressurization after RCIC failure, and prevention of repressurization by holding open the SRVs (including consideration for normal leakage)?

2. Weather in northern Taiwan is hot and humid for approximately 6 months in a year. After restoration of AC power and air-conditioning following an extended SBO event, is there any possibility of moisture condensation leading to electrical faulting of critical equipment (e.g., condensate dripping from the ventilation duct into electrical cabinets or onto electrical devices), based on the Lungmen design?

3. Figure 19.3-1 indicates that at least two curves (temperature and pressure) should be included in the figure but only one is shown. In addition, there appear to be inappropriate labels on the figure and abscissa. Why is vessel pressure not included in Figure 19.3-1? The last paragraph in Section 19.3.1.2.2 is somewhat confusing and its inclusion in the text seems unnecessary.

4. What are the estimated SBO frequencies and SBO core damage frequencies for various external events?

PSAR Response:

1. The purpose of manual relief operation is to reduce SRV cycling while maintaining enough steam pressure for RCIC operation until recovery of off-site power, one of the EDGs, or the Swing EDG. To estimate an upper bound for the number of manual SRV open/close cycles, a bounding SAFER analysis was used. The analysis was performed for an ABWR with similar sized SRVs assuming a Main Steam Line (MSL) break from 102% RTP. In the analysis the MSIVs auto closed after 3.0 seconds into the event. One or two SRVs cycled in the safety mode a total of 14 times within 700 seconds of the start of the transient. The SRV cycling was terminated by RCIC and decreasing reactor pressure below the lowest SRV setpoint. The reactor operator is expected to manually open the SRV to maintain the reactor pressure below the high reactor pressure scram setpoint. Depending on the decay heat in the reactor, the operator can elect to close the SRV at lower reactor pressures to minimize the SRV operation. Because of the wider band of reactor pressure control by the manual operation of the SRV, it is expected that the number of manual SRV cycles would be less than 14 (bounding case) in combination with RCIC injecting cold water into the reactor.
   As indicated in the response to Question 19-022, the capacity of the nitrogen bottles is large enough to provide 16 times the nitrogen needed to open the 8 ADS SRVs, and after the ADS valves are open, to supply enough nitrogen to substitute for 7-day leakage from the valves accumulators. It is concluded that there is a sufficient nitrogen capacity for RPV pressure control during an SBO.

2. The raceway and enclosure design includes features to minimize the entrance of moisture, including condensation, into enclosures and to minimize the impact of moisture that may enter. Design features are investigated and used as practical, such as gasketed enclosures, avoidance of top entry of conduit, drip loops in conduit, weep holes drilled in cabinet bottoms, and seals in conduit. The restoration period following an extended SBO event is a severe design condition and will be considered in the equipment and raceway design to minimize impacts. Electrical protection is also provided by fuses or circuit breakers to remove electrical faults before significant equipment or cable damage can occur.

3. For questions on Figure 19.3-1, please see response to part 1 of Question 19-001. Please see Figure AJ.2-6a, page AJ.2-54, Attachment AJ of PSAR Appendix A, for the RPV pressure for the SBO accident. As for the last paragraph in Section 19.3.1.2.2, the intent is to describe the behavior of the transient during the first 6 hours in those figures where behavior during this period has been truncated. We agree that the paragraph could be confusing and will be replaced by a more clear and concise statement in the revised PSAR.

4. SBO CDF is 2.55E-6 /reactor year for seismic events, 7.95E-9 /reactor year for fire, 1.5E-10 / reactor year for typhoon, and is negligible for internal flooding.
   As explained in the response to Question 19-025, item 3, the frequency of SBO (with and without core damage) due to external events is not currently available and will be calculated for the revised PSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-027

PSAR Sections: 19.3.1.3

Question Date: March 2, 1998

PSAR Question:

1. Given inadvertent actuation of the automatic fire suppression system, how will the Lungmen NPS be designed to reduce the likelihood of equipment damage caused by electrical faulting resulting from water spray and subsequent dripping of water onto electrical devices?

2. In the currently operating nuclear plants, hot shorts induced by fires could result in adverse impact on the ability of the plant to respond to the event. Hot shorts in control cables can cause repositioning of valves, spurious operation of equipment leading to component damage, LOCAs, and other undesirable equipment operation, etc. In instrumentation circuits, hot shorts may cause misleading displays potentially leading to inappropriate control actions. How will the Lungmen NPS be designed to minimize the extent of the adverse conditions caused by fire-induced hot shorts?

3. A fire in an area shared between the two adjacent units might cause a simultaneous trip demand and impact equipment on both units. Are there any locations in the Lungmen NPS that are shared between the two units; e.g. control room?

4. What are the estimated fire initiation frequencies in the most risk-significant plant location in terms of fire compartments? What are the fire-induced core damage frequencies associated with these locations?

PSAR Response:

1. Control and electrical equipment is contained in enclosures which provide some protection from water spray. Additionally, the raceway design includes features to prevent the ingress of water to a cabinet or panel. Section 9.5.1.1.7, Spurious Control Actions, addresses this issue further.

2. Control circuits are protected from short circuits by fault detection circuitry, fuses, or circuit breakers. It is highly unlikely that a control or instrumentation circuit could suffer limited damage so that it would short to another conductor, cause an undesired action or indication, and yet remain in service without blowing a fuse or tripping a circuit breaker. For further reference please see Section 9.5.1.3.10, "Electrical Cable Fire Protection"; Section 9.5.1.3.11, "Fire Separation for Safe Shutdown"; and Section 8.3.3.6, "Independent Redundant Systems".

3. The Swing Diesel Generator is shared between the two units and is located in the Auxiliary Fuel Building (AFB). The Swing DG is a backup for three divisional EDGs serving each unit. The Swing DG is designed in accordance with IEEE 384 separation methods to preclude the possibility of common mode failure disabling more than one DG during a fire in the AFB. The shared hot machine shop, radwaste building, service water pump house, circulating water pump house, and related tunnels are similarly designed to preclude the possibility of a fire-induced common cause failure causing a simultaneous trip demand or impacting safety equipment on both units.

4. Please see page AD.3-1, Attachment AD of the PSAR Appendix A, for the fire initiation frequency in the fire risk-significant plant locations and Table AD.1-1, page AD.1-3, for the corresponding core damage frequency due to fire.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **19-028**

PSAR Sections: **19.3.1.4**

Question Date: **March 2, 1998**

PSAR Question:

1. Resolution of the ISLOCA issue requires that low pressure piping systems interfacing with the reactor coolant pressure boundary (RCPB) be designed to withstand reactor pressure to the extent practicable. For some low pressure systems attached to the RCPB, it may not be practicable or necessary to provide a higher system ultimate pressure capability for the entire low pressure connected system. The systems interfacing with the RCPB that have not been designed to withstand the full reactor pressure should include (1) the capability for leak testing the pressure isolation valves, (2) indication in the control room of valve position when isolation valve operators are deenergized, and (3) high pressure alarm to warn control room operators when rising reactor pressure approaches the design pressure of attached low pressure systems or when both isolation valves are not closed. Please provide information to show which of these additional protection features will be included in the Lungmen NPS design for each of the applicable systems that have been determined to interface directly or indirectly with the RCPB.

2. In Section 19.3.1.4.1, one of the design requirements listed for satisfactory ISLOCA protection is that "the design pressure for the low-pressure piping systems that interface with the RCPB should be equal to 0.4 times the normal operating RCPB pressure of 7.07 MPaG." This implies that the ultimate capacity of the interfacing piping is 2.5 times the design pressure. For some interfacing LOCA scenarios, the interfacing system's piping and components may be exposed to temperatures greater than those experienced during their normal operating modes. How are temperature effects, if any, taken into account for interfacing systems' piping exposed to primary coolant at normal operating temperature? How are uncertainties in the ultimate pressure capacity taken into account?

3. Based on the Lungmen NPS design, what are the estimated frequencies of challenges to the low pressure systems by the full reactor pressure due to failures at the various high/low pressure interfaces? Considering the upgrades to the pressure capacity incorporated in selected Lungmen low pressure systems, what is the estimated frequency of ISLOCA leading to a direct release to outside of the primary containment? What are the CDF contributions from the various potential pathways?

PSAR Response:

1. Please see Appendix 3M and 3MA of PSAR Chapter 3 for a detailed discussion of the low pressure interfacing systems design provisions to prevent ISLOCA. In particular, Section 3M.3, "Boundary Limits of URS," discusses the protection provided for those systems where it is impractical to upgrade to URS.

2. As stated in Section 19.3.1.4.1, the USNRC has defined four design requirements (as well as periodic surveillance and leak testing). The fourth design requirement specifies that the design is to be in accordance with ASME BPV Code Section III, Subarticle NC/ND-3600. This subarticle prescribes the proper temperature to use in the analysis.

3. The frequency of ISLOCA and LOCA outside the containment are estimated to extremely low as explained in the PSAR Appendix A, page A1-14. Because of their low frequencies relative to other LOCAs, and the minor contribution of LOCA to the core damage frequency, ~2% of the internal initiating events CDF, the ISLOCA and LOCA outside the containment were not explicitly considered as significant initiators in the PRA of Appendix A.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **19-029**

PSAR Sections: **19.4.1**

Question Date: **March 2, 1998**

PSAR Question:

1. Although the access tunnels are addressed in Section 19.4.1, there is no discussion relative to how these tunnels are isolated from the lower drywell region during normal operation or what role, if any, these tunnels play in severe accidents. Assuming that hatches are provided for such isolation, have their pressure capacities been evaluated? Please provide some additional details in this regard.

2. On page 19.4-2, it is noted that "during plant startup, inerting of the PCV is initiated at least 24 hours prior to the plant's reaching 15% power." However, no reference is made to the deinerting procedure during shutdown of the plant. What are the requirements imposed for the shutdown process?

3. On page 19.4-3, it is noted that the Flammability Control System is manually operated if hydrogen is present. Has the potential for operator error induced deinertion and/or ignition been addressed?

4. On page 19.4-3, it is also noted that nitrogen is added to the COPS discharge piping by opening the COPS purge supply and exhaust valves then adding nitrogen via pressurized nitrogen bottles. The outboard rupture disc has a very low setpoint. How is rupture of the outboard disc avoided during the vent line inertion procedure? Is instrumentation provided to indicate the outboard rupture disc is intact subsequent to the addition of nitrogen?

PSAR Response:

1. The purpose of Section 19.4.1 is to provide an overview of the containment design. Access tunnels and their airlocks and closures are part of part of the primary containment boundary, and their integrity is

maintained in accordance with PSAR Section 16.3.6. Detailed analysis and discussion of severe accidents and mitigative features including the containment are included in Sections 19.3 through 19.7. In particular, Section 19.7 will address the containment structural capability and leakage through penetrations during severe accidents. Please see PSAR Chapters 3 and 6 for detailed design description, structural design, and design basis accident analysis.

2. Please see response to Question 19-011

3. Deinerting the containment is controlled by technical specifications, (PSAR Section 16.3.6.3.2) and plant procedures. It is allowed only for no more than 24 hours before (shutdown) or after (startup) 15% RTP is reached. FCS is designed to recombine hydrogen and oxygen generated by metal-water reaction during LOCA. Although manually initiated, the LOCA initiation signal from the main control room is used to start the system. The chance of an operator error that will deinert the containment especially with hydrogen present in excess of the flammability limit is extremely remote.

4. Inerting the COPS discharge piping uses nitrogen at pressure below the set point of the outboard rupture disc. The nitrogen pressure will be controlled and monitored during inerting to ensure that the pressure in the discharge piping is within specification limits. The specific test procedure to ensure that the disc is intact after inerting will be defined in the FSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-030

PSAR Sections: 19.4.2

Question Date: March 2, 1998

PSAR Question:

In Section 19.4.2, it appears MAAP is the basic code employed for severe accident progression analyses. However, the discussion from page 19.4-11 on contains numerous references to the BWRSAR code and a few references to the MELCOR code. Please clarify which codes were used for the various parts of the analysis.

PSAR Response:

The MAAP-ABWR code is the code used by GE for severe accident analysis of the ABWR. The BWRSAR and MELCOR codes have been used by USNRC consultants to validate the MAAP-ABWR analysis. The recommended clarification will be included in the revised PSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-031

PSAR Sections: 19.4.3.1

Question Date: March 2, 1998

PSAR Question:

1. Section 19.4.3.1.1 discusses clad oxidation and hydrogen generation for MAAP runs in which the blockage and eutectic cutoff models are disabled. Is it correct to assume that all references to blockage refers to the MAAP 3.0B "channel blockage model?" While the assumption of no blockage is conservative with respect to hydrogen productions, it may be nonconservative with respect to fission product retention. Was the impact on source terms assessed for both the no blockage and local blockage cases?

2. Throughout Chapter 19, the units of pressure are stated as MPaA, MPaG, or Mpa. It is not always obvious that Mpa is used only for differential pressure. For example, in Section 19.4.3.1.2 (page 19.4-7), it is stated that "consideration of 100% fuel clad metal-water reaction results in a peak pressure of about 0.618 MPa." Is this an absolute or gauge pressure? This same comment applies to the pressure capability goal listed on the next page as well as throughout Chapter 19. Is there an implied default unit when the suffix A or G is not provided?

Response:

1. The analysis in Chapter 19 used the ABWR version of MAAP3.0B. It is true that the MAAP3.0B channel blockage control parameter has been set to the value that disables the blockage and eutectic cutoff models. The following tables present key fission products release parameters for the four cases discussed in Section 19.4.3.1.1. Cases 1 and 2 in Table 1 refer to low pressure core melt scenarios with debris cooling in the containment by ACIWA spray and LDF, respectively. Cases 3 and 4 (Table 2) are high pressure core melt scenarios with debris cooling in the containment by a combined LDF/ACIWA spray and LDF, respectively. Except for Case 4, where absence of containment spray leads to leakage

through upper drywell penetrations before COPS rupture disk opens, radioactive release in the other cases is through COPS after scrubbing in the suppression pool and occurs when the wetwell airspace pressure increases to the rupture disk set point.

The MAAP3.0B-ABWR results for the four cases of Tables 1 and 2 below indicate that the assumption of no blockage increases the source term. As seen in these tables, disabling the blockage and eutectic cutoff in the MAAP analysis results in an earlier RPV failure (compared to the blockage case or Base Case) which results in an earlier containment challenge. The earlier containment challenge leads in turn to an earlier release from the containment through COPS (Cases 1 through 3) or upper drywell penetrations (Case 4). The earlier containment release provides less time for radioactivity decay and aerosol settling within the containment, and consequently a higher radioactivity level of released fission gases, and higher release fractions of radioactive aerosols . This is illustrated by the CsI release fractions with and without blockage shown in the following tables for Cases 1 and 4. The MAAP results of Cases 2 and 3 also show increase in the release fractions when the blockage and eutectic cutoff is disabled in the MAAP analysis. However, the increase in these latter cases is not large enough to increase the CsI release fractions above 1.E-7.

2. For consistency with the practice utilized during the US ABWR Certification Project, an A, G, or D will be applied to the pressure units to indicate absolute, gauge, or differential pressure.
In the PSAR Chapter 19, all MAAP code output pressures and unqualified pressures are in absolute units.

Table 1 - Low Pressure Melt Cases

Parameter	Case 1 Base Case	Case 1 Block/eutectic Disabled	Case 2 Base Case	Case 2 Block/eutectic Disabled
Zr Oxidized (%)	6.3	15.8	6.3	15.8
RPV Failure Time (hr)	1.8	1.1	1.8	1.1
Rupture Disk Open Time (hr)	31.1	30.6	20.2	16.7
CsI Release Fract at 72 hrs	~1E-7	~1E-6	<1E-7	<1E-7

Table 2 - High Pressure Melt Cases

Parameter	Case 3 Base Case	Case 3 Block/eutectic Disabled	Case 4* Base Case	Case 4* Block/eutectic Disabled
Zr Oxidized (%)	5.1	35.9	5.1	35.9
RPV Failure Time (hr)	2.0	1.8	2.0	1.8
Rupture Disk Open Time (hr)	25.0	19.7	18.1**	7.1**
CsI Release Frac. at 72 hrs	<1E-7	<1E-7	8.7E-2	12.5E-2

* Case assumes no containment spray, which leads to leakage through upper drywell penetrations.

** Start time for Release through drywell penetrations.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-032

PSAR Sections: 19.4.3.2

Question Date: March 2, 1998

PSAR Question:

1. In Section 19.4.3.2.1.4, it is stated that the use of a low gas content (basaltic) "translates into a long time to pressurize the containment." This type of concrete, however, typically ablates faster than other types of concrete leading to faster times to melt-through to the embedded liner. Were any sensitivity studies performed using different types of concrete for the sacrificial layer in the lower drywell? If so, what were the results and conclusions of these studies relative to overpressure failure and liner melt-through? It is noted that the impact of the type of concrete selected for the pedestal is discussed in Section 19.4.3.2.2.7.

2. Section 19.4.3.2.1.6 does not address the capability of the Containment Overpressurization Protection System (COPS) relative to pressure spikes (i.e. sudden pressurization). For example, is there sufficient vent area to limit peak pressures that might exceed the COPS setpoint? A brief discussion of this capability, or lack thereof, should be addressed in this section along with a reference to more detailed information.

3. The decomposition event tree concept is introduced in Section 19.4.3.2.2.1 with essentially no explanation. Has the reader been introduced to this concept earlier in the PSAR? There should be a brief introductory explanation as to the purpose of these trees and how they relate to traditional containment event trees.

4. In the discussion of the decomposition event tree for debris coolability in Section 19.4.3.2.2.1, no reference to the possibility that the debris might not spread uniformly over the entire lower drywell floor area could be found. How have uncertainties in the drywell spread area been addressed?

5. For the discussion of Case 3 on page 19.4-18, the argument does not appear to support the assignment of probabilities that is identified. Would not a split of 0.5 for "No CCI" and 0.5 for "Wet CCI" be more appropriate?

6. For the decomposition event tree for pedestal resistance to CCI which is discussed in Section 19.4.3.2.2.2, what is the rationale for the ordering of events? Event 4 (ratio of radial to axial erosion) appears to have no prior dependencies but Event 2 (suppression pool water floods lower drywell after downcomer penetration) would appear to be dependent on the degree of radial erosion.

7. In the evaluation of pedestal strength in Section 19.4.3.2.2.3, it appears as though only axial compressive stresses are considered. Are there any significant stresses due to bending?

Response:

1. No sensitivity studies using different types of concrete for the sacrificial layer in the lower drywell were performed. The fact that basaltic concrete ablates faster than other types of concrete is recognized in the MAAP-ABWR calculations. MAAP calculations of most likely severe accident scenarios confirm that the sacrificial bed has an adequate depth to protect the containment liner. In effect, the faster ablation rate is compensated for by the basaltic concrete depth.

2. A discussion will be added to Section 19.4.3.2.1.6 on the question of pressure spikes. The discussion will cover the likelihood and severity of such spikes and COPS design basis. The above discussion will be included in the revised PSAR.

3. An introduction to the Decomposition Event Trees (DET) will be added to Section 19.4.3.2.2.1. The introduction will describe the DET logic, event types and their quantification, and DET relation to the CET. The above introduction will be included in the revised PSAR.

4. The impact of non-uniform distribution of the debris has been excluded from discussion because it was judged to have second order effect. A discussion will be added to Section 19.4.3.2.2.1 to explain the basis for

this conclusion. The discussion will identify potential causes and configurations of non-uniform debris distributions, and the probability and effect of such distributions. The above discussion will be included in the revised PSAR.

5. The basis for Case 3 probabilities may be explained as follows. As stated in page 19.4-18, the range of upward heat flux for this case is from 200 to 400 kW/m^2 It is convenient to divide this range into two parts:
   1) From 300 - 400 kW/m^2. This range is similar to that of Case 2. As indicated in page 19.4-18, the probability of "No CCI" for Case 2 is 1.0.
   2) From 200 - 300 kW/m^2. This range falls between Case 2 and Case 4. Therefore, the probability of "No CCI" for this range is between 0.0 and 1.0. A value of 0.5 for the "No CCI" probability has been chosen as an unbiased value. .

   Assuming equal probabilities (0.5) for Case 3 to be in one of the above two ranges, the resulting "No CCI" probability is estimated as 0.5 x 1.0 + 0.5 x 0.5 = 0.75. The "Wet CCI" probability is 1.0 - 0.75 = 0.25. These are the probabilities assigned for Case 3 in page 19.4-18.

   We find the suggested split of 0.5 for "No CCI" and 0.5 for "Wet CCI" to have merit and will be evaluated in sensitivity analysis planned for the FSAR.

6. Radial erosion by CCI up to the pedestal vertical channels (25 cm) presents an insignicant threat to the structural integrity of the pedestal. The decomposition event tree of Figure 19.4-4 applies for CCI erosion > 25 cm. This allows the possibility of suppression pool water flow to the lower drywell, and debris flow to the suppression pool. The first three events in the decomposition event tree are related to the issue of debris coolability and account for the above flow between the lower drywell and the suppression pool. This determines the extent of axial erosion. The events are followed by the ratio of radial to axial erosion since radial erosion impacts the pedestal integrity or pedestal failure which is the last event in the tree.

7. Please see Section AJ.12.4 of Appendix A for the basis of the strength evaluation approach.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-033

PSAR Sections: 19.4.3.3

Question Date: March 2, 1998

PSAR Question:

1. In the discussion of the fraction of entrained debris fragmented and transported to the upper drywell at the bottom of page 19.4-36, it is noted that "a 50/50 split would occur based on equal flow areas in both directions." Are the lengths of these pathways also equal?

2. In NUREG/CR-6338 (Ref. 1), which assessed the direct containment heating issue for Westinghouse PWRs with large dry or subatmospheric containments, load distributions were convoluted with containment strength distributions to determine the conditional containment failure to probability. This approach has also been used in a number of PRA and IPE studies to assess containment failure probability for rapid pressure increases. Based on a review of Section 19.4.3.3.2.2 and Figure 19.4-18, it appears as though this approach was not used in the Lungmen analysis. The approach used for Lungmen appears to ignore the possibility of containment failure at less than the median failure pressure. How might the conditional probabilities of drywell failure discussed in Sections 19.4.3.3.2.2 through 19.4.3.3.2.4 be impacted if the stress-strength interference approach is applied? What impact does COPS have on pressure spikes?

PSAR Response:

1. The 50/50 split choice is an engineering judgment that accounted for the area and other factors such as length and geometry. As stated in page 19.4-35, the gas transport pathway to the upper drywell is relatively convoluted and the impacted debris is likely to flow downward toward the wetwell vents. As indicated in page 19.4-36 the 50/50 split value is believed to be conservative.

2. The conditional probabilities of drywell failure in Section 19.4.3.3 are

the same as those developed in the Standard ABWR SSAR. The SSAR containment failure conditional probabilities have been derived using a stress-strength approach. Please see Section AJ.18 of Appendix A for the containment strength distribution. As for the second part of the question, COPS will relieve the drywell pressure if the wetwell airspace pressure reaches the COPS rupture disk set point.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-034

PSAR Sections: 19.4.3.4

Question Date: March 2, 1998

PSAR Question:

1. In the last paragraph on page 19.4-40, reference is made to "moveable" penetrations. What are these penetrations?

2. How were pressure spikes considered in the COPS pressure setpoint determination (Section 19.4.3.4.1? What was the basis for selecting the configuration in which the rupture disc is normally exposed to containment pressure as contrasted to a situation in which the rupture disc and vent line is normally isolated and venting is controlled by the operator? What offsite dose impacts result from opening of the containment rupture discs?

3. In Section 19.4.3.4.5, it is stated that relatively small time differences of 2 to 4 hours will not significantly affect the magnitude of the offsite dose. While this may be true, won't the additional time be important in emergency response actions, and hence risk?

PSAR Response:

1. "movable", or "operable" penetrations refer to non-fixed mechanical or electrical penetrations such as the drywell head closure, equipment hatches, and personnel airlocks.

2. Please see response to Question 19-032, part 2 on the issue of pressure spikes. The COPS concept was chosen because it is passive and requires no operator action to relieve the containment pressure. Once this relief is obtained, the containment integrity can be recovered by the COPS two isolation valves. Please see Table 19.4-3 for the effectiveness of COPS in reducing the radioactive release.

3. The time elapsed from the severe accident initiation to radioactive release

via COPS is on the order of 20 to 24 hours for most of the accident scenarios analyzed. The addition of 2 to 4 hours to this elapsed time will not have a significant impact on the probability of successful recovery action.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-035

PSAR Sections: 19.4.3.6

Question Date: March 2, 1998

PSAR Question:

1. On pages 19.3-11 and 19.4-57, reference is made to a Firewater Addition System. This system did not appear to be addressed in Section 19.1 or 19.2. In addition, a Firewater System is cited in Section 19.4.3.6.5.2.4. Are the Firewater Addition System, Firewater System, and the ACIWA System one in the same?

2. In the second sentence of the fourth paragraph of Section 19.4.3.6.3.2, "impulse loading" or "impulse pressure" is missing. It is also noted that "the pressure experienced by the pedestal wall will be reduced because the shock wave has to pass through some amount of water before it impinges on the wall." Won't the existence of a free surface also diminish the shock wave as it passes through the water?

3. In the calculation of T (the natural period of the pedestal) on page 19.4-65, all material properties used in the formula are for steel except for density which is for concrete. What is the justification for using concrete density in this application?

PSAR Response:

1. The "Firewater Addition System", "Firewater System", and the "ACIWA" refer to the same function which is formally referred to as the AC Independent Water Addition System or ACIWA. The use of "Firewater" is derived from the fact that water from the Fire Protection System is used by the ACIWA.

2. The second sentence will be changed to read "The impulse pressure in Figure 19.4-22 is conservative because ...." We agree that the free surface will diminish the shock wave as it passes through the water.

3 As stated in page 19.4-65, the determination of the various parameters of the composite pedestal structure can be very complicated. The use of concrete density is a simplifying approximation for the composite structure because of the large relative concrete mass in the structure.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-036

PSAR Sections: 19.4.3.7

Question Date: March 2, 1998

PSAR Question:

1. Figures 19.4-25a through 19.4-25c show essentially no increase in drywell pressure and temperature for the accident scenario (100% clad oxidation) discussed in Section 19.4.3.7.1.1.1. Furthermore, the peak containment pressure indicated in these figures is less than 0.4 MPa whereas in Section 19.4.3.1.2 (p. 19.4-7), a value of 0.62 MPa was reported. Are these results consistent with one another? Why is no increase in pressure and temperature apparent in Figure 19.4-25? Why are the peak pressures different in these two sections?

2. In the first paragraph on page 19.4-80 (last sentence), it is stated that "heat transfer in the long pipe runs allows the process fluid to remain within survivability limits." What are the survivability limits being referenced?

3. In Item (10) on page 19.4-83, reference is made to operable penetrations. Are these the same as the "movable" penetrations alluded to in Question 12? A discussion of the various types of penetrations should be provided in Section 19.4.1, Overview of the Containment Design.

PSAR Response:

1. The containment pressure of 0.62 MPa of Section 19.4.3.1.2 is based on non-mechanistic, conservative, design basis calculations which assume a design basis LOCA combined with 100% metal water reaction at the same time. The drywell pressure of Figure 19.4-25a was obtained by MAAP code analysis. Although the analysis included conservative modeling assumptions to produce 100% metal water reaction, the results are more realistic in that they considered a TMI-type (isolation) accident with multiple operator actions to turn ECCS on and off to obtain the 100% metal water reaction. As stated in the second paragraph of Section

19.4.3.7.1.1.1, page 19.4-74, a multiplier that effectively increased the rate of metal water reaction was used in the MAAP model. This led to the rapid increase of the containment pressure from ~0.1 to ~0.4 MPa which corresponds to the oxidation of 100% of the Zr cladding. This explains why the pressure remains almost constant (notice that the unit of the time axis is in hours).

2. The "survivability limits" are pressure and temperature limits for the RHR components (piping, valves, ...etc.) which are discussed in pages 19.4-80 and 81.

3. Please see response to Question 19-034 part 1 for the definition of movable and operable penetrations. The recommended discussion in Section 19.4.1 will be included in the revised PSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-037

PSAR Sections: 19.4.3.8

Question Date: March 6, 1998

PSAR Question:

In Section 19.4.3.8.2.1.5.2.1, a formula is given for the change in energy due to additional superheat. The results of applying this formula are given in Table 19.4-13. As shown in Table 19.4-9, both the debris specific heat and the latent heat of fusion are functions of temperature. Were these temperature dependencies factored in to the calculations leading up to the results tabulated in Table 19.4-9?

PSAR Response:

The difference in the specific heat and the latent heat in Table 19.4-9 is due to differences in the debris composition. The properties have been derived using the debris material composition for each scenario shown in Table 19.4-9 and the properties of Table 19.4-10 which are temperature independent.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

RECOMMENDATIONS/TYPOS

PSAR Sections: 19.4.3.8

Question Date: March 6, 1998

1. Throughout Chapter 19, a number of typos were found. This section lists some of the typos identified. It is suggested that a thorough editing of this chapter be performed and a revision to this Chapter be issued.

2. The pressure head capacity of the ACIWA should be identified in Section 19.2 or 19.3.

3. In the first sentence of Section 19.3.1.2.1.1, it is stated that "the RCIC provides water to the reactor vessel to assume core cooling during a station blackout...." Should the word "assume" be "resume?"

4. In Section 19.4.1, various containment dimensions are provided but no information is provided relative to the free volumes of the various containment regions. The identification of these free volumes in this particular section would appear to be appropriate.

5. On page 19.4-4 (line 7), it is stated that the Lower Drywell Flooding System (LDF) is a subsection of the CMS (i.e., the Containment Monitoring System). The passive LDF does not appear to be functionally similar to the CMS. Subsequent paragraphs indicate that perhaps "CMS" should have been "CCS". Is the latter the correct citation?

6. Relative to the four sensitivity studies discussed in Section 19.4.3.1.1, a tabulated summary of the results for the four sensitivity cases as well as the cases with the blockage and eutectic cutoff models not disables would be useful. The accident scenarios preceding the time of ACIWA initiation should also be described.

7. In the first paragraph of Section 19.4.3.6.2.3, Figure 19.4-21 is incorrectly cited. The correct citation is Figure 19.4-20.

8. In the fourth paragraph on page 19.4-66, Figure 19.4-23 is incorrectly cited. The correct citation is Figure 19.4-23.

9. In Section 19.4.3.2.1.1, it is stated that "the walls of the floor drain sump shield have channels which permit water flow, but which will not permit debris flow." A brief explanation as to why water flow is possible but debris flow is not possible should be included at this point of the text with a reference to the more detailed analysis found in Section 19.4.3.8.2.

10. In Section 19.4.3.6.3.4.2, it is stated that the smallest impulse load expected to fail the pedestal is 0.024 MPa. the correct units should be MPa scc.

11. Figure 19.4-11 appears to have the split fractions on the wrong branches (i.e., nothing is provided for the intermediate containment pressure case for which this figure applies)

PSAR Response:

1. All the recommendations will be included in the revised PSAR except as follows:
   #3. The word "assume" will be changed to "assure"
   #5. The LDF is a subsystem of the CMS as shown in PSAR Figure 6.2-46 (Sheet 2 of 5)
   #8. The Figure number at the end of the first paragraph and the fourth paragraph in page 19.4-66 will be corrected to read "Figure 19.4-22"

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： A-001

問題章節(PSAR Section)： Appendix A- overview

初提日期(Question Date)： 1997.12.22

問題內容(PSAR Question)：

1. 10CFR100 requests the two conditions to be fulfilled.
   (1) An individual located on the exclusion area boundary for two hours immediately following the onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.
   (2) An individual, located on the LPZ(Low Population Zone) boundary during the entire period of radioactive cloud resulting from the postulated fission product release, would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

2. The TPC bid specification requires that the PRA analyses include internal and external events (earthquake, typhoon, fire, internal flood, tsunami, etc.). However, the tsunami is not analyzed in the PRA. Please explain.

3. Plant changes could potentially introduce new initiating events or result in previously screened out events becoming more important. Please show the systematical process to this issue.

4. Population center distance means the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,00 residents (10CFR100.3(C)). As depicted in figure2.1-4. Keelung is the nearest one. Please show statistics that no other areas are closer to Lungmen.

問題答覆(Responses)：

1. The Lungmen Nuclear Power Station (NPS) meets both deterministic licensing requirements, such as those specified in of 10 CFR 100, and probabilistic requirements specified in the TPC Bid Specifications. The scope of PSAR Appendix A is confined to probabilistic risk analysis to confirm that the Lungmen NPS meets the specified

probabilistic requirements. Please see Chapter 15 for discussion of Lungmen NPS compliance with 10CFR100 requirements. Some of the key differences between the scope of deterministic and probabilistic analyses to meet the above requirements are noted below.

The deterministic requirements of 10CFR100 apply to "Design basis" accidents (DBAs). In contrast, PRA covers a spectrum of low probability accidents which are more severe than DBAs. The PRA also covers more consequences and radiation-exposure locations than those specified in 10CRF100. Specifically, the PRA consequences include short term and long term health effects due to radioactive release for individuals residing at different directions and radial distances from the plant. The health effects are derived from estimated doses to the whole body, thyroid, as well as other organs. The doses in turn are calculated for the spectrum of identified low probability severe accidents using the spectrum of weather conditions appropriate for the plant site. Please refer to PSAR Appendix AL for PRA consequences calculations.

2. Section A6.6, Other External Events, presents external events that were evaluated qualitatively only and why they were not considered for detailed quantitative analysis. Tsunami is discussed in page A6-12, where it is shown that the maximum flood level will be below the plant grade level and therefore tsunami need not be considered for detailed PRA quantification.

3. Figure A1-1 of the PSAR Appendix A presents an overview of the methodology used to develop the Lungmen PRA. The same methodology and the Lungmen baseline PRA data and analysis results will be used as a basis for evaluating the impact of plant changes.

   As seen in Figure A1-1, the approach starts by qualitative evaluation. Such an evaluation will indicate whether the plant change is significant enough for detailed quantitative analysis, or its impact is beneficial or insignificant based on simple bounding or qualitative analysis. The "Plant Familiarization" task will relate the design change to the baseline PRA system(s), structure(s), or component(s) (SSC). Other qualitative and quantitative tasks shown in Figure A1-1 provide a checklist of the issues that must be addressed for qualitative evaluation. A quick walkdown through all of these tasks should be performed to identify tasks that need closer evaluation and those that can be safely ignored. To ensure that such an evaluation is both reliable and efficient, maximum use should be made of the Lungmen baseline PRA data and analysis results, and insights of the proper experts (PRA, system engineers, licensing, O&M).

A key feature of the Lungmen baseline PRA is the discrete and hierarchical structure of the results of the various tasks shown in Figure A1-1. The fault tree of Figure A4-1, for example, starts at the top by dividing the initiating events (IE) into local and global ones. Each is then divided into IE categories. Finally, the IE categories are related to SSC-related events, specific external events, ...etc. Similarly, Table A11-1, presents the core damage categories and the SSC-related accident sequences contributing to each category. The dependency matrices of Section A3.6 also provide a convenient format for assessing the potential dependencies of the plant changes.

The results of the above walkdown will be locating the specific IE, accident sequences, core damage categories, containment event sequences, and radioactive release categories that may be significantly impacted by the plant change. They will also include new or increased vulnerabilities to external events or dependent failures that may result from the change.

Given the above results, the importance of the change can be evaluated by relative comparison of its contribution to the frequency of IE categories, core damage categories and radioactive release categories. The change in the risk can then be estimated using the contribution of the above categories to the risk.

If such a qualitative evaluation leads to satisfactory results, with the concurrence of the proper experts, the results will be reported. Otherwise, additional selective modeling and requantification of the baseline PRA will need to be performed.

4. a) Please refer to attached Figure-1. The figure shows that the distance from the nearest boundary of Keelung city to the Yenliao site is more than 15 km, which is far more than the distance of one and one-third times the low population zone. (300 meters x 4/3 = 400 meters).
   b) From attached Figure-2, all population centers other than Keelung city are more than 20 km away from the Yenliao site.

No PSAR revision is proposed in response to this question.

<u>**ROCAEC's further review comments:**</u>

1. What is the data collection period in calculation of the maximum flood level of tsunami ?
2. What is the reference plant in your PRA? Are there initiating events added in or

deleted from the reference plant PRA? Please list the added/deleted events.

3. Please provide the electronic file for all the Lungmen PRA input data (such as I.E. frequencies, Basic Event Values, Seismic fragility,...etc.)
4. According to PSAR Appendix AK there is no Lungmen specific radioactive release categories, how can you compare the design change of SSCs contribution to the frequency of release categories? What will be the impact of release categories for SSE increase from 0.3g to 0.4g?

Further Clarification:

1. 本公司核四廠最大可能海嘯及暴潮之評估係委託國立成功大學台南水工試驗所進行研究，其報告乃參考美國核能管制委員會 Standard Review Plan 中 sec. 2.4.5 (Probable maximum surge and seiche flooding) 及 sec. 2.4.6 (Probable maximum tsunami flooding) 之海嘯及暴潮規範，並採用美國國家標準局 (ANSI) 海嘯研究小組建議的方法從事研究。
   為了估計核能四廠附近可能引起最大海嘯之最大可能地震規模，該研究報告中檢視了自 1901 年 1 月至 1983 年 6 月之間實際發生過的地震，並以所發生過之最大芮氏地震規模再加 0.2 作為將來可能發生之最大地震規模。
   至於，颱風則採用 1932 年至 1983 年共五十二年較完整之颱風氣象資料，在所設定的範圍內（北緯 15°~28°，東經 121°~132°）分別予以逐年逐月選出發生於北太平洋西部海面上之颱風中心最低氣壓值。

2. The reference plant modeled in the PRA is the Lungmen NPS design described in the Lungmen PSAR. The US Standard ABWR, which has been certified by the USNRC, is similar, but not identical, to the reference plant. Please see response to Track number A-017, item 3, for a discussion of key similarities and differences between the two ABWR designs.

   The significant similarities between key design features and design ground rules of the Lungmen and US Standard ABWR plants made it possible for the Lungmen PRA to benefit from the significant data base and licensing experience of the Standard ABWR and use this information where applicable. On the other hand, the Lungmen PRA includes site-specific seismic PRA and typhoon PRA, which are not included in the US Standard ABWR PRA reported in the SSAR. Table A-001a-1 provides a comparison of the internal and external initiating events used in the two PRAs.

3. Electronic files for component data used in the CAFTA code for internal events Level 1 analysis, and fragility data used in the seismic PRA are attached.

<< LNG_INT1.XLS >> Internal PRA fault trees basic events probability
<< LUNGFG82.XLS >> Seismic PRA components fragility

4. The following overview of the Lungmen PRA approach for calculating the frequency of release categories will help in explaining how design differences are factored in the PRA. Tables AL-8 and AL-9 of PSAR Appendix A define the release parameters of 15 release categories which were used in the Lungmen PRA consequence analysis. The second column of Table AL-7, page AL-14, presents the accident sequences contributing to each of the above 15 categories. Each sequence is formed of 8 characters[1] which represent the accident type that led to the core damage (first four characters), mitigative features that operated (next two characters), containment release mode (next character), and magnitude of the release (last character). For example, the first sequence of Case 1 in Table AL-7 (LCHPFSRN) refers to an accident involving loss of core cooling with vessel failure at high pressure (LCHP), ACIWA fire water spray operating (FS), release through COPS after rupture disk ruptures (R), and the radioactive release is dominantly noble gases (N: <100% noble gases, <0.1% volatiles). Please see Section AJ.2 for the definition of these characters.

   The frequency of any of the above sequences can be expressed as the sum over the range of initiating events of the product of the following terms:

   (1) Frequency of the initiating event
   (2) Conditional probability of an accident type, e.g., LCHP, given the initiating event
   (3) Conditional probability of mitigative features operation, containment release mode, and magnitude of release, e.g., FSRN, given the intiating event and accident type.

   The first term above depends on the plant design for internal events and some external events such as fires and internal flooding, and also on the plant site for external events such as seismic and typhoon. As indicated in the attached Table A-001a-1, the Lungmen PSAR PRA used the Yen-Liao site hazard curves for seismic and typhoon events.

   The second term above is obtained from accident sequence event trees and fault trees which represent the design and operator actions for the specific plant analyzed. For

[1] A few 9-character sequences ending with "D90" appear in the table. These sequences present the low probability event of drywell failure before the rupture disk design pressure of 0.72 MPa (90 psig) is reached.

seismic events, component fragilities are used to estimate the conditional probability of components failure. These fragilities depend, among other factors, on whether the component is seismically qualified or not, the component strength margin to meet the SSE design requirement, soil-structure interaction (SSI), ...etc. Please see PSAR Section AC.3 for the factors considered in the seismic capacity analysis in the Lungmen seismic PRA.

The third term in the above product may represent events which are random, e.g., random failure of the ACIWA (therefore FS does not appear in the sequence), events induced by the initiating event, e.g., seismically induced failure of the RHR heat exchanger with subsequent suppression pool draining to the RHR room, or accident-induced event, e.g., activation of the passive flooder when the temperature increases in the lower drywell. Containment event trees are used for estimating the conditional probability of the third term.

As seen from the above discussion, design changes may impact any of the above three items. For this reason, evaluation of the impact of design changes on the risk requires a walk-through the PRA approach as explained in the response to Track # A-001. It should be noted, however, that the change in the SSE from 0.3g to 0.4g does not necessarily mean design change. For example, if a component designed for a 0.3g has a significantly large seismic safety factor because of considerations other than seismic loadings, then the same component may be qualified for 0.4g or larger SSE.

Table A-001a-1

Comparison of PRA Initiating Events Used in the Lungmen PSAR and the US Standard ABWR SSAR

Operating Mode	Initiating Event Type	Initiating Event	SSAR	PSAR
Power	Internal	- Manual Shutdown	yes	yes
		- Isolation/Loss of Feedwater		
		- MSIV Closure	yes	yes
		- Loss of Cond. Vac.	yes	yes
		- Press. Reg./Bypass	yes	yes
		Valve Closed	yes	yes
		- Loss of Feedwater	yes	yes
		- Non-Isolation		
		- Turbine Trip with	yes	yes
		Bypass	yes	yes
		- IORV	yes	yes
		- Loss of Off-site Power	yes	yes
		- Small LOCA	yes	yes
		- Medium LOCA	yes	yes
		- Large LOCA		
	External	- Seismic	no (1)	yes (2)
		- Fire		
		- Three Safety Divisions	yes	yes
		- Control Building	yes	yes
		- Turbine Building	yes	yes
		- Switchgear Building	no	yes
		- Internal Flooding		
		- Control Building	yes	yes
		- Reactor Building	yes	yes
		- Turbine Building	yes	yes
		- Tornado	yes (3)	no
		- Typhoon	no	yes (2)
Shutdown	Internal	- Loss of RHR	yes	yes
		- Loss of Off-site Power	yes	yes
		- Loss of Service Water	yes	yes
		- Draindown	no	yes
		- LOCA	no	yes
		- Loss of AC Bus	no	yes
	External	- Seismic	no	yes
		- Fire	yes (4)	yes
		- Internal Flooding	yes (4)	yes

(1) Seismic margin assessment only. Not site-specific

(2) Yen-Liao-site specific

(3) Non-site specific

(4) Qualitative assessment was performed to evaluate the design defense-in-depth capability

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-002

PSAR Sections: A1.3

Question Date: December 9, 1997

PSAR Question:

In Table A1-6, the top rank of event "FASTRAN-Fast Transfer to Main Generator" is assumed to be 0.5 of probability for operator error. Please provide the technical base for such an assumption and briefly explain why this value is considered to be conservative.

PSAR Response:

In the current Lungmen design, the normal preferred off-site power supply and the main generator provide 29 kv power to three unit auxiliary transformers (UATs). The UATs supply medium voltage power to the Class 1E and non-class 1E distribution buses. The event "FASTRAN" refers to the unavailability of UAT power from the main generator given loss of off-site power.

The main generator is connected to the UATs through the normally closed generator breaker. On loss of offsite power, the switch yard breaker opens and the main generator continues to provide power to the UATs, as long as the turbine provides the needed power and the generator breaker remains closed. This is accomplished without automatic or manual transfer between buses. The balance between reactor power production and house loads is achieved by the load following capability of the plant, where turbine steam control (by throttling and turbine bypass) is used to prevent turbine over-speed and reactivity control (and not scram) is used to reduce the nuclear power to the appropriate level. Turbine trip, or failure of a UAT or the generator breaker will lead to the FASTRAN event.

Several BWRs in Europe and Japan have the capability to accept load rejection incidents without scram. The GE-Leibstadt BWR6 design has shown an excellent scram-avoidance reliability record, where power from the main generator has been available to run house loads in all known tests and incidents involving loss of off-site power. No data is currently available on other BWR experience other than the knowledge that initial plant startups were successful. Due to the small data base and lack of detailed unavailability analysis at this time, the value of 0.5 was assigned for the "FASTRAN" event. The value is believed conservative based on the positive BWR operation and startup experience indicated above.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-003

PSAR Sections: Appendix AA

Question Date: December 9, 1997

PSAR Question:

Please describe the differences between pre-accident and post-accident human errors. why the screen values have 2 orders of differences?

PSAR Response:

Appendix AH of the PSAR contains detailed analysis of the pre-accident and post-accident human error probabilities (HEPs) used in the PRA. Tables AH.2-1 and AH.3-1 summarize the obtained results. As seen from these tables, both pre-accident and post-accident HEP values cover a broad range (from 10^{-5} to 10^{-3} for pre-accident HEPs and from 10^{-6} to 1.0 for post-accident HEPs.) The tables also show that pre-accident human actions are generally routine and simple, e.g., instrument calibration, while post-accident actions cover a broad range of complexities.

The above wide ranges of values are not uncommon in HEPs used in PRAs. It has long been recognized that, HEP estimates do vary by several orders of magnitudes even among seemingly similar actions. Such variations can be attributed to various factors including differences in:

1) Clarity and completeness of emergency and maintenance procedures
2) Accessibility and unambiguity of information required to determine what action to take.
3) The specific steps to be taken to complete the action
4) Complexity of the above steps
5) Operator training and familiarity with the action
6) Extent to which action is independently verified or validated
7) Time allowed for the action
8) Stress level
9) Dependence of the action success on plant conditions and prior actions
10) HEP Modeling and basic data

In recognition of the above factors, the Lungmen PRA devoted a significant effort to analyze human reliability and to document the procedures, assumptions, and data used in the HEP quantification in Appendix AH.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-004

PSAR Sections: Appendix AJ.2

Question Date: January 20, 1998

PSAR Question:

In the LCHP-PS-R-N sequence,

(a) What is the fraction of debris that may fall in the lower vessel head before vessel breach?

(b) What is the fraction of the corium are assumed to be carried to the drywell and wetwell when vessel breach occurs? How is the fraction determined?

Response:

(a) The UO2 mass in the RPV, upper and lower drywells, and the wetwell is shown as a function of time for this scenario in Figure AJ.2.3e, page AJ.2-45. The fraction of debris in the lower vessel head before vessel breach is 53% of the core.

(b) The corium fractions carried to the drywell and wetwell when vessel breach occurs are 86% and 14% of the ejected corium (53% of the core), respectively. The drywell/wetwell split fraction has been assigned by judgment based on insights from the direct containment heating model of Section AJ.10.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-005

PSAR Sections: Appendix AJ.3.1

Question Date: January 20, 1998

PSAR Question:

1. In section AJ.3.1.4, the triggering condition of steam explosion with RPV depressurization was demonstrated. However, the steam explosion under high RPV breaching pressure is very much concerned.

(a) Please discuss the possibility and triggering condition for steam explosion under high RPV breaching pressure.

(b) Does the containment maintain its integrity under this condition?

2.(a) In Table AJ.3-1, the mass flow rate of corium from vessel is500 kg/s. Please explain how this value is obtained.

(b) In Section AJ.11.6.1.1 (p. AJ.11-19), it is mentioned that "The maximum rate of debris ejection from the vessel is about 6000 kg/s." Please use that flow rate to re-calculate AJ.3.1.4 and to determine the possibility of steam explosion in ABWR.

(c) In section AJ.11.6.2.1.1, it is mentioned that "The bound areaof the low plenum vessel failure is 0.1 m^2." In Table AJ.3.1, Q(volumetric flow rate of corium from vessel) = 0.056 m^3/s. If A = 0.1 m^2, then v = 0.56 m/s. This result quite differs fromthat in AJ.3.1.4 (the debris stream velocity = 11 m/s). Please explain the difference.

Response:

1.

(a) As concluded in Section AJ.11.1.1, the total frequency of accidents involving core damage and a pre-flooded lower drywell is less than 4.5E-10 per year. Since core damage frequency in the ABWR is dominated by low pressure accident sequences, the frequency of a high pressure melt and a pre-flooded lower drywell is even lower than the above frequency. Despite such a low frequency, a scoping conservative calculation using the approach of Section AJ.3.1.3 and the ABWR application of Section AJ.3.1.4 was completed to demonstrate that the necessary conditions for steam explosion are not satisfied for the most likely high pressure melt ejection accident sequences as summarized below.

Section AJ.10.2 indicates that the most likely HPE accident will involve a small vessel failure area ($< 0.1\ m^2$) and a small fraction of the core (~ 10% or 24,000 Kg). The SRVs in the ABWR are designed for pressure relief before the RPV pressure reaches 9.5 MPa. The acceleration of the above mass under the maximum pressure is estimated as follows:

Acceleration
$= 9.5\ MPa \times (101{,}972\ Kgf/m^2 / MPa) \times (9.81\ m/sec^2) \times 0.1\ m^2 /$
$24000\ Kg$
$= 37.93\ m/sec^2$

The above acceleration leads to a drag acceleration (Equation AJ.3.-4) of ~ $(37.93 + 9.81) / 9.81 = 4.87$ times the drag acceleration of the low pressure case.

Using Equation AJ.3-3, the stable droplet radius for the high pressure case is estimated to be larger than the square root of (1/ 4.87) or ~ 0.45 the radius of the low pressure case. From Equation AJ.3-12, the time constant τ_h is proportional to the droplet radius. Using the value of τ_h of 9.2 s estimated in page AJ.3-11 leads to the follwoing high pressure time constant τ_h (HP):

τ_h (HP) > 9.2 x 0.45 = 4.3 s

Therefore, the necessary condition for steam explosion of Equation AJ.3-29 is not satisfied. This result provides confidence that steam explosion is not possible for the most likely high pressure melt ejection accidents.

(b) Although the above scoping calculation shows that steam explosion is not possible for the high pressure melt ejection, the analysis in Section AJ.11 shows the impact of a bounding steam explosion does not present a threat to the containment. Please refer to Section AJ.11 for details on this bounding analysis.

2

(a) The value of 500 kg/sec is an average flow rate for the initial release based on MAAP analysis. (Please see Figure AJ.11-9 for the range of initial values).

(b) A scoping calculation similar to the one used in the response to Question 1 (a) above leads to the conclusion that the time constant τ_h in this case can be given by:

τ_h (6,000 Kg/s) > 9.2 x 500 / 6,000 = 0.77 s

Therefore, the necessary condition for steam explosion of Equation AJ.3-29 is not satisfied. Consequently, no steam explosion is expected for this case.

(c) The speed of Section AJ.3.1.4 is estimated at the surface of the water pool which is 6 m below the bottom of the reactor vessel according to Table AJ.3-1. (Ignoring the initial speed, the speed at the pool surface can be estimated as $v = \sqrt{2 g H} = \sqrt{2 \times 9.81 \times 6} = 10.85 \sim 11$ m/sec). Naturally, this is different from the speed of the corium as it leaves the RPV.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-006

PSAR Sections: AJ. 16 RCCV Nonlinear Analysis

Question Date: January 19, 1998

PSAR Question:

1. Please provide major input data of parameters necessary for the non-linear analysis.

2. Which computer code (and why) was used in the non-linear analysis? What are the element types used and numbers of iteration cycles for non-linear computation?

3. Please provide the locations of maximum stresses in Table AJ.16-1.

Response:

1. As stated in the first paragraph of page AJ.1-1, Attachment AJ of PSAR Appendix A contains analysis that was performed for the US Standard ABWR. Prior to the certification of the Standard ABWR, the USNRC has reviewed the SSAR containment structural capacity analysis and validated the results by performing independent analysis as discussed in the USNRC Final Safety Evaluation Report (FSER), Section 19.2.6, pages 19-70 through 19-75. Lungmen-specific input data will be prepared on completion of the containment design and compared to those used for the Standard ABWR to ensure that the Lungmen containment has the same structural capability when compared to that of the Standard ABWR. The above evaluation will be reported in the FSAR.

2. Please see response to part 1 above.

3. Please see response to part 1 above.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-007

PSAR Sections: **AJ.17.1.1 Concrete Shell**

Question Date: **January 19, 1998**

PSAR Question:

It is mentioned that the pressure buildup rate within the containment is longer than one second. What the fastest pressure buildup rate and the associated accident?

Response:

There are three types of accident phenomena that can result in relatively fast pressure buildup: hydrogen detonation, steam explosion, and direct containment heating (DCH) resulting from high pressure melt ejection. As discussed in PSAR Chapter 19, the ABWR has protective features against each these mechanisms. These features include containment inerting which prevents hydrogen burning or detonation, incoherent melting process inside the vessel and unavailability of water in the lower drywell which prevent steam explosion, and RPV depressurization by the ADS which prevents high pressure melt ejection. Moreover, Section AJ.11 contains an analysis of an extremely unlikely bounding fuel-coolant interaction scenario in the lower drywell and shows that such an extreme steam explosion will not challenge the containment integrity. Direct containment heating is discussed in Section AJ.10 where it is shown that the risk from DCH is also insignificant.

Apart from the above risk insignificant accidents, the severe accidents analyzed by MAAP for the ABWR show a drywell pressure buildup rate of less than 0.2 MPa / hr. as the pressure approaches its peak value.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-008

PSAR Sections: Appendix AE, AG, AJ

Question Date: January 3, 1998

PSAR Question:

1. The flooding analysis discusses the capabilities of the US ABWR Standard Plant to withstand internal flooding (e.g., service water, suppression pool line breaks.) The report does not address the feasibility to Lungmen plant (Attachment AE).

2. In thermal hydraulic calculations, it is clearly depicted that no fuel damage will occur as long as the fuel remains covered by water. However, does the coverage include two-phase water in success criteria?

3. Please clarify the success criteria used for core integrity among such terms as core damage, core uncovery, 2/3 core height in the thermal hydraulic analysis.

Response:

1. As described in Section A1.1.2, the first step in the Lungmen PRA approach was to assess the applicability of the Standard ABWR PRA to the Lungmen NPS. Investigation of the SSAR flood PRA indicated that its use for Lungmen was appropriate based on the following observations:
   1) The accident sequences contributing most to the flood risk are dominated by human error, flooding detection, failure to close watertight doors between Control Building and Access Control Building, and failure to provide adequate core cooling.
   2) The insights gained from the SSAR flood PRA have been incorporated in the Standard ABWR design.
   3) The Standard ABWR and Lungmen water sources, flood detection and mitigation features, operator actions, and floor arrangements are basically identical.

Consequently, except for seismically-induced flood, where Lungmen NPS is designed for 0.4g SSE vs. 0.3g for the Standard ABWR, the flood risk for Lungmen NPS is expected to be similar to that of the Standard ABWR.

Based on the above conclusions, Lungmen-specific flood PRA was performed only for seismic events. Please see Section AC.8 for the summary of Lungmen seismically-induced flood PRA.

2. In the thermal hydraulic calculations, fuel coverage by water provides a sufficient, not a necessary, condition for no fuel damage. As described in Question 3 below, two phase coverage of the core will be sufficient to prevent core damage provided that the collapsed water level is at 2/3 the core height or higher. This provides a more restrictive sufficient condition for core damage prevention. Please see response to Question number 3 for the necessary condition for core damage.

3. Shutdown accidents that lead to core damage involve RPV water level drop. Core uncovery starts when the water level drops to the Top of Active Fuel (TAF). Core damage is considered to occur when the fuel cladding is breached, leading to gap release of the fission gas. As described in the last paragraph of page AG.4-11, Sandia National Laboratories calculations for Grand Gulf indicate that BWR gap release does not occurs until a few hours after the collapsed water level reaches TAF. Using the Sandia results, the Lungmen shutdown PRA developed the following simple relationship to estimate the time to gap release.

   Time from shutdown accident intiation to gap release =
   Time (from shutdown accident intiation) for the collapsed water level to drop to 2/3 core height + δT,

   where δT is the time to gap release, measured from the time at which the collapsed water level reaches 2/3 core height. The value of δ**T** depends on the decay heat level at the time the water level reaches the 2/3 core height. Based on Sandia 's results, δT is estimated to be 0.5 hour if the water level drops to 2/3 core height 1 day after shutdown. The time to core damage (i. e. gap release) is estimated for other times after shutdown by Equation AG.4-19, page AG.4-12.

The time for the water level to reach 2/3 core height is estimated in the shutdown PRA assuming an adiabatic heat up. The calculation starts by estimating the time required to raise the RPV water temperature to the saturation temperature, then add the time required to evaporate the water initially above the 2/3 core height to the point which corresponds to 2/3 core height, and the time to gap release, δT. Please see Tables AG.4-3 and AG.4-4 for estimates of the time to the start of core uncovery (water level reaching TAF), time to the 2/3 core height level, and time to core damage for shutdown accidents postulated to occur 2 days and 30 days after reactor shutdown.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-009

PSAR Sections: Overview, A

Question Date: January 3, 1998

PSAR Question:

1. According to 10CFR50.34(f)(1)(vii), the applicant is required to perform a feasibility and risk assessment study. The purpose of this study is to determine the optimum automatic depressurization system (ADS) design modifications that would eliminate the need for manual activation to ensure adequate core cooling. This is, however, not clearly shown in the PSAR. Please explain.

2. The safety goal of societal risk and individual risk are both set to 0.1%. Both goals are considered relative to risk from other causes. Please provide the relative data used for the calculations.

3. GE claimed to make a number of design modifications to the ABWR both early in the design and later during NRC review of the ABWR PRA that were motivated by the results of the PRA. Please provide list of the major modifications made to the design for Lungmen.

Response:

1. The intent of 10CFR50.34(f)(1)(vii) is to provide more assurance of adequate core cooling in the event of transients and accidents not producing a high drywell pressure signal under conditions where high pressure makeup systems are unable to maintain reactor inventory.

   In response to the above item, General Electric and the BWR Owners' Group participated in a program to evaluate potential modification of the ADS initiation logic. The program investigated the feasibility of automating the ADS for isolation events with and without a stuck open relief valve, and assessed the changes in the overall plant risk resulting from such automation. The program identified and evaluated five options and the results were submitted to the USNRC in March 1981. The

USNRC judged that two of the options are acceptable and either option may be used. The two options are:

1) Eliminate the high pressure trip from the ADS initiation logic and add manual inhibit switch, or
2) Bypass the high drywell pressure trip after runout of a timer started at low pressure ECCS initiation level and add manual inhibit switch.

In the certified ABWR, the second option has been chosen where an 8 minute high drywell pressure bypass timer has been added to the ADS initiation logic. This timer will initiate on a Low Water Level 1 signal. When it times out, it bypasses the need for a high drywell pressure signal to initiate the standard ADS initiation logic. For all LOCAs inside the containment, a high drywell pressure signal will be present and the ADS will actuate 29 seconds after a Low Water Level 1 is reached. Please see Chapter 1, Appendix 1A; "Response to TMI Related Matters," Section 1A.2.26; "Modification of Automatic Depressurization System Logic - Feasibility for Increased Diversity for Some Event Sequences," for a more complete discussion of the bypass logic.

2. Risk of cancer fatality per person per year (due to all causes) is based on the USA statistics reported in Reference 1. The value used for PRA calculations is 2.0E-3 per person per year. This leads to a societal risk goal of 2.0E-3 x 0.1/100 = 2.E-6 cancer fatalities per reactor year.

   Individual risk of accidental fatality per person per year (due to all causes) is based on the USA statistics reported in Reference 2. The value used for PRA calculations is 3.9E-4 per person per year. This leads to an individual risk goal of 3.9E-4 x 0.1/100 = 3.9E-7 early fatalities per reactor year.

   References:

   1. "1986 Cancer Facts & Figures," American Cancer Society, USA, 1986
   2. "Accident Facts - 1988 Edition," National Safety Council, USA, 1988

3. The Standard ABWR final design is the product of several years of optimization effort to meet both qualitative risk reduction goals such as

design simplification and quantitative risk reduction goals. Specifically, in the early phase of the design, PRA was used to guide design decisions in the following areas:

1) Core cooling systems
2) Reactivity control reliability
3) Instrumentation reliability
4) Control rod design improvement
5) RIP simultaneous trip prevention

As part of the ABWR certification effort, the PRA was further used to improve the design. This effort resulted in the addition of the following mitigative features:

1) AC-Independent Water Addition (ACIWA)
2) Combustion Turbine Generator (CTG) as an alternate electric ac power supply
3) Lower Drywell Flooder
4) Containment Overpressure Protection System (COPS)

Please see SSAR Section 19.7 for detailed description of the above and other improvements in the Standard ABWR.

The above improvement led to an optimized design which has been used as a basis for the Lungmen NPS with three main differences;

1) Designing the Lungmen NPS turbine with increased steam bypass capability in order to accept a load rejection incident without scram
2) Adding a Safety Class Swing DG as an alternate ac source (The certified ABWR CTG has been removed.)
3) Designing the Lungmen NPS for a higher SSE (0.4g as opposed to 0.3g for the certified ABWR).

Robustness of the ABWR design was demonstrated in the Lungmen NPS internal events PRA by importance analysis and sensitivity analysis which did not uncover any particular design weakness. However, the seismic PRA identified components requiring enhancement of their capacity beyond the generic values. This is understandable, since the Lungmen

NPS requirements specify quantitative risk goals which include contributions from internal as well as external events (Standard ABWR quantitative risk goals are specified for internal events only). Table AC.3-10 of PSAR Appendix A, Attachment AC, identifies the components requiring seismic capacity enhancement and how the capacity may be improved. Please see Section AC.10 for a detailed discussion of the seismic risk results and insights from the seismic PRA.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-010

PSAR Sections: Appendix AB

Question Date: February 18, 1998

PSAR Question:

1. In page A6-4 (section A6.2.3.1) of seismic PRA at power, the Figure A6-1 is mistyped as Figure A6-2.

2. Figure A6-2 is missing (current Figure A6-2 is actually Figure A6-3), please resubmit this figure.

3. In Figure AB.5-3 and Figure A11-3, the legends for IB and ID are obviously reversed, please adjust them.

Response:

1. Figure A6-1 has been relabled as Figure A6-2 and missing Figure A6-1 has been added as shown in the attached figures.

2. Figure A6-2 has been relabled as Figure A6-3.

3. Errors in Figures AB.5-3 and A11-3 have been corrected as shown in the attached markups of these figures.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-011

PSAR Sections: Appendix A6, AD

Question Date: February 18, 1998

PSAR Question:

The titles of Fig. AD.4-7 and AD.4-8 are wrong. Please correct them.

Response:

The titles of Figures AD.4-7 and AD.4-8 have been corrected as shown in the attached markups.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-012

PSAR Sections: Appendix A13, A14, AB, AG, AJ, AL

Question Date: November 28, 1997

PSAR Question:

Figures AJ. 15-2, AJ. 16-2, AJ. 16-3 have inconsistent units and errors. Please correct them.

Response:

The PSAR Attachment AJ does not have Figures AJ.16-2 or AJ.16-3. A review of Sections AJ.15, AJ.16, and AJ.17 revealed inconsistent units or errors in the following figures: AJ.15-1, AJ.15-2, AJ.16-1, AJ.17-1, and AJ.17-5. These inconsistencies have been corrected as shown in the markups of these figures.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-013

PSAR Sections: Appendix A6.2, AC

Question Date: February 18, 1998

PSAR Question:

1. For seismic PRA analysis, the top five accident sequences listed in section A6.2.3.2.1 is inconsistent with section A1.4.1.2 and AC.10.2.1, please justify.

2. The CDF (core damage frequency) of SBO (station blackout) and ATWS (anticipated transient without scram) are very much incorrect. Please justify.

3. Please explain why all sequence frequencies for seismic event tree of Figure AC.6-1 (sheet 1 through 6) are set to zero.

4. For CDF distribution over seismic hazard curve (Figure A1-11, A6-3, AC.10-3), please explain why group 0.4-0.6g has lower CDF than the group 0.0-0.4g. What are CDF values for those five groups?

Response:

1. Section A6.2.3.2.1 has not been updated to reflect the latest results of the Lungmen seismic analysis. Section A6.2.3.2.1 has been corrected to be consistent with Sections A1.5.1.2 and AC.10.2.1 as shown in the attached markup.

2. As indicated in Item 1 above, Section A6.2.3.2.1 has not been updated resulting in the incorrect values for the SBO and ATWS percent contribution to the seismic CDF. This error has been corrected as shown in the attached markups.

3. The sequence frequencies column will be deleted from Figure AC.6-1 Sheets 1 through 6. Since transfer to some of these event trees comes

from more than one event sequence (e.g., transfer to Sheet 4 from sequences number 10 and 11 of Sheet 1), it was found preferable to present the sequence frequencies in tabular form with proper reference to Figure AC.6-1 sheet number. The attached table A-013-1 shows the annual frequencies of the sequences and their relation to Figure AC.6-1.

4. The CDF contribution of each ground acceleration range depends on three factors: the frequency of a seismic event within the specified ground acceleration range, random failure of the components in the accident sequences leading to core damage, and seismically-induced failure of these components within the given ground acceleration range. Over the range of ground acceleration 0.0 - 0.4g, the only significant seismically-induced failure is the loss of off-site power (median capacity 0.3g). Random failure of components is the main contributor to the CDF. In the range of 0.4 - 0.6g, random failure is also the dominant failure mode. Since the frequency of seismic events within this range (1.264E-3 /year) is much smaller than in the range from 0.0 - 0.4g (9.106E-1 /year), the CDF contributions from the seismically induced loss of off-site power and random failure are smaller. This explains the lower CDF frequency over the range 0.4 - 0.6g relative to that of the range 0.0 - 0.4g. For ground accelerations greater than 0.6g, contributions to the CDF from seismically-induced failures become significant. This explains the increase in the CDF contribution beyond 0.6g. The seismic CDF contribution redeclines in the range of 1.0 - 1.2g because the decline in the seismic frequency is more significant than the increase in seismic-induced failure probabilities in this range. The CDF contribution of seismic events within the five ranges is shown in the following table.

Ground Acceleration Range, g	CDF
0.0-0.4	1.33E-07
0.4-0.6	6.87E-09
0.6-0.8	3.89E-07
0.8-1.0	1.54E-06
1.0-1.2	1.06E-06

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-014

PSAR Sections: Appendix A, Attachment AC

Question Date: November 28, 1997

PSAR Question:

1. Current result shows 1.8g is the lowest fragility value for Reactor internal. Please verify that the fragility value for RIP (reactor internal pump) would be higher than 1.8g.

2. Some components capacities have been raised as shown in Table AC.3-10. Are all of these components within GE's scope? Is GE responsible for the increased fragility value of fire protection water pump (from 1.8g to 2.8g)?

3. Please include all components appeared in seismic fault tree cutsets in Table AC.3-11 "Seismic capacity summary". For example, swing DG, fire protection water pump.

4. Has plant-specific SSI analysis been completed yet? If yes, please reflect the result to the fragility (i.e., F_{SA} or F_{SSI}) calculation. If not, when will the SSI analysis be done?

Response:

1. The failure mode of interest for the reactor internals is structural deformation that may prevent full insertion of the control rods. The relatively low median value of 1.8g is, in part, a result of this failure mode definition. Since the RIP does not interfere with the control rod insertion, the failure mode of interest is different. Specifically, since the RIP forms a part of the reactor coolant boundary, seismically-induced leakage of the primary coolant is a more appropriate failure mode. However, the RIP capacity against leakage should be the same as that of the RPV since it is a part of its boundary. According to Table AC.3-11, the RPV seismic median capacity is 5g which is much larger than 1.8g.

2. All of the components in Table AC.3-10 are within GE scope. The fire protection water pump is included in Table AC.3-11 and not AC.3-10. As seen in Table AC.3-11, last line in page AC.3-26, the fire water pump median capacity used in the seismic PRA is the generic value of 1.9g.

3. Table AC.3-11 has been revised to include all the component categories. Markups of the table are attached. Table A-014-1 below shows the correspondence of Table AC.3-11 and the fault tree components used to quantify the seismic event tree models.

4. Plant-specific SSI analyses for Seismic Category I buildings have not yet been completed. The plant-specific fragility calculations will be provided in the FSAR.

Table A-014-1 Relation of Seismic Fault Tree Events and Table AC.3-11

MODEL DESIGNATOR	TABLE AC.3-11 CATEGORY
R_BLDG	Reactor Building
CONTNMNT	Containment
PEDESTAL	RPV Pedestal
CTR_BLDG	Control Building
RPV_ANC	Reactor Pressure Vessel
SHROUD	Shroud Support
CRD_GD	CRD Guide Tubes
CRD_HS	CRD Housing
FUEL_ASS	Fuel Assemblies
CRD_HYCU	Hydraulic Control Unit
AC_TRAY	Cable Trays
DC_TRAY	Cable Trays-critical; DC trays
CST TNK_FW TNK1_FW	Large flat-bottom storage tanks (CST, ACIWA Tank)
HX_SW HX_ECW	Heat Exchanger (SW)
HX_RHR	Heat Exchanger (RHR)
OP_XFORM	Off-site Power
BAT_RACK	Batteries and battery racks
INVT	Inverters
CIRT_BRK MCC R10MCCSEFAIL E11MOVSEISOL	Switchgear/Motor control center
XFORM	Transformers (480V)
DG DG7	Diesel generators & support systems
PUMP_UR	Turbine-driven pumps (RCIC)
PUMP_SW	Motor-driven pumps (SW)

MODEL DESIGNATOR	TABLE AC.3-11 CATEGORY
PUMP_UH PUMP_SLC PUMP_V PUMP_RCW PUMP_ECW	Motor-driven pumps (HPCF, LPCF, SLC)
PUMP_FW	Fire Water Pump
TNK_SLC	Small tanks (SLC)
VLV_MOSW VLV_MOSS VLV_MODS VLV_MOMF VLV_INJ VLV_MOINJ VLV_MFW VLV_FMIJ VLV_SLC VLV_RCW VLV_ECW	Motor-operated valves
VLV_CKSW VLV_SRV VLV_CHK VLV_CKDS	Safety relief and check valves
AC_DUCT	HVAC ducting
ROOM_ACU	Air handling units/Room A.C.
PIPE_SW PIPE_UR PIPE_UH PIPE_V PIPE_RHR PIPE_CNT PIPE_SLC PIPE_RCW PIPE_ECW	Piping
PIPE_FW	Buried piping
PUMPH_FW	Fire water pump house
PUPH_SW	Service water pump house

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-015

PSAR Sections: Appendix A1, A4

Question Date: November 28, 1997

PSAR Question:

1. Please provide MCSs of important sequence, up to 90% of CDF contributors. They should be listed sequence by sequence, including all modes of operation (power and shutdown, internal and external).

2. Please provide detail calculation or reasoning sheets for those "negligible" special initiators listed on Table A4-2.

Response:

1. The MCSs of important sequences, contributing to 90% of the internal events CDF at power and 90% of the shutdown CDF are attached. The top accident sequences contributing to 90% of the seismic CDF during power, fire CDF, internal flooding CDF, and typhoon CDF are also attached.

2. Detailed discussion of the reason for concluding that the special initiators of Table A4-2 result in negligible risk contribution will be provided in the PSAR Amendment.

ROCAEC Review Comment:

1. As the most important contributor, Seismic PRA contributes up to 90% of the total CDF. In the attachment, only sequence information (which can be easily found in the PSAR) is provided, but not the useful MCS information. Without MC, review cannot be continued. We need GE's commitment on when can this information be provided to reviewers.
2. We expect to see the detail explanation in the Amendment.

Further Clarification:

The ISAP computer code that GE used for seismic analysis does not provide the requested minimum cutsets (MCS). The calculation logic of the code is confined to the resolution of event tree sequences which we sent to the ROCAEC. To obtain minimum cutsets which include fault tree basic events (similar to those obtained by the CAFTA Code for internal events) will require a significant effort either to change the ISAP Code calculation logic, or to expand the event trees to the basic events level. Either of these options will increase the number of accident sequences exponentially. We recognize also that the ISAP Code logic does not include the calculation of

CAFTA-like importance measures.
As a part of our plans for the FSAR, we will identify and evaluate options to deal with the above ISAP Code shortcomings later this year. The best option will be implemented for the FSAR calculations.

Although the ISAP code does not provide minimum cut sets, the output includes the contributions to each event tree node from the corresponding fault tree basic events. These contributions provide an insight into the relative importance of basic events to the event tree node probability. We think this insight can serve as a realistic substitute for the insight gained from minimum cutsets. We can develop an adequate description of the ISAP output and send it along with the code output to the ROCAEC. Although this can not be provided in time for Amendment I, we will provide it as supplemental information before the end of 1998.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-016

PSAR Sections: Appendix A6, AD

Question Date: November 28, 1997

PSAR Question:

1. Fire Core Damage Frequency shown in Table AD.1-1 are different from those in Table AD.4-2 and A1-10. Please explain. There are two different results in different places of the report, e.g. paragraph A.1.4.2, A.6.3.2, AD.1.3.

2. Please provide more detail information about how to get the fire compartment ignition frequency in page AD.3-1. Calculation sheets are preferred.

3. Please add CDF data on the last column of Fig. AD.4-1 through AD.4-6. The information currently available is not enough to review and check the results.

Response:

1. Table AD.1-1represents the most recent Lungmen fire PRA results. We regret that the other tables and parts of the text have not been properly updated. These errors have been corrected as shown in the attached markups.

2. The fire compartment ignition frequency has been derived from the detailed evaluation reported in Appendix 19M of the SSAR. Specifically, adjustments to the calculations in the SSAR have been made to reflect differences between the standard ABWR and Lungmen design features that could significantly affect fire frequencies, e.g., use of non-qualified cable, and fire frequency trends in operating nuclear power plants.

   The calculation sheets and SSAR tables used in the calculation of the Lungmen fire ignition frequencies are attached. The following summary identifies the area affected, the change in the Lungmen design implemented in estimating the Lungmen fire ignition frequencies, and, the basis for the change in the fire frequency.

Area Affected :	Turbine Building
Change :	Remove Offgas systems and non-qualified cables.
Basis:	Lungmen does not use non-qualified cables. Offgas system fires have trended down from a very high frequency to a very low frequency since the mid 1980s. Credit fire watch for welding fires: Industry experience indicates that a continuous fire watch will substantially reduce the chance of a significant fire. (The value 0.028 is a typical factor used in many studies submitted by US utilities to the USNRC for GL 88-20 Supplement 4.)

Area Affected :	Divisional areas
Change:	Remove diesels and non-qualified cables
Basis:	Lungmen does not use non-qualified cable. Diesels are in a separate area. Credit fire watch for welding fires. See Turbine Building discussion for welding fires.

Area Affected :	Control Room
Change:	Remove all but electrical panels
Basis for Change:	The original SSAR analysis applied the EPRI FIVE method to the Control Room in a manner contrary to its intended application. Control rooms have not experienced fires for reasons other than electrical panels. The other causes, the largest of which is welding, have never occurred in a control room. Using the SSAR fire frequency, we would have expected to see approximately 60 to 70 more control room fires than we have seen in the approximately 2000 reactor-yrs of US experience.

3. The CDF data has been added to the last column of Figures AD.4-1 through AD.4-6 of the attached markups.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-017

PSAR Sections: Appendix A13, A14, AB, AG, AJ, AL

Question Date: November 28, 1997

PSAR Question:

1. It was mentioned in Section A13.1.1 that CET of external events are described in Attachments AC, AD, AE and AF. It was found that the CET for Fire (AD), Flood (AE) and Typhoon (AF) are described in specific subsections but the one on Seismic (AC) is missing. Please supplement this part of the information.

2. Section A14.1 (3) indicated that "The elevation of the release is conservatively assumed to be at the reactor building roof height" which is not consistent with the information TPC provided to AEC in the November 14, 1997 meeting. Please clarify.

3. Section A14.6 explained that the Source Term and Containment release Analysis described in Attachment AJ were all based on US Standard ABWR and the major changes for Lungmen such as SSE goes from 0.3g to 0.4g, etc. were not taken into account. Brief discussions of the impacts of these changes should be provided and also the schedule for modification should be planned.

4. It was pointed out in Section AB.6.11.3.3.2 that the probability of recovery of high pressure injection system was described in detail in subsection A.J.4.2 but no quantitative results in that subsection was provided to support the data in this section.

5. Section AG.8.4.1.8 declared that Shutdown CET for subclass VIA similar to that for internal CET for class ID but the probabilities referenced were not correct. For instance, in class ID, the RHR recovery probability for core melt not arrested and no active injection to lower D/W is "0" but in subclass VIA it was "0.8". (Ref. pages AB-6-38 and AG-8-13)

6. Please explain whether MAAP 3.0B-ABWR code has been reviewed and approved by USNRC ? (Section AJ.1.1.2)

7. The assumptions used for the present analysis (such as Limestone sand concrete used for containment structure, Basaltic used for lower drywell flow, and RCIC room cooling can sustain 8 hours, etc.) should have a schedule for their verification. (Section AJ.1.2.1)

Response:

1. The CETs used in the seismic PRA are attached. The first event in the CETs is the core damage category. The annual frequency of these categories were estimated by the ISAP Code. The probabilities of other events shown in the CETs were estimated using the approach described in the Proposal seismic PRA, Section C.5. A section that explains the CETs and the basis for the probability values used will be included in the revised PSAR.

2. The second sentence of A14.1(3), page A14-2, indicates that two release elevations are used in the PRA consequence analysis; reactor building roof height, and ground level. As explained in item (3) of Section A14.1, conservative rules are used for selecting one of these release elevations depending on the expected release location. The use of two elevations instead of only one conservative elevation, such as the ground level used in the design basis analysis, is consistent with the PRA groundrule of providing realistic, though conservative, analysis.

3. Tables 1.3.1 through 1.3.4 of the PSAR provide a comparison between the design characteristics of the Lungmen NPS and the certified US Standard ABWR. The tables show two primary areas where the two ABWRs are different; fuel design and performance, and seismic design basis. A brief discussion of these differences is given below. This is followed by a discussion of differences not included in the above tables which have been accounted for in the PSAR PRA, and the general plan to ensure that the PRA adequately reflects the risk of the as-built Lungmen NPS.

   The Lungmen NPS will use the GE-12 fuel type. Key differences between this fuel and the GE-8 type assumed for the certified ABWR design are shown in PSAR Table 1.3.1. It is particularly noted that the Lungmen fuel has the following characteristics:

- Lower maximum and average volumetric fuel temperature (500 degrees F lower than the Standard ABWR) which means that it will take longer time for core heatup to melting.
- Smaller Zr mass in the fuel cladding and fuel assembly ducts, which means less hydrogen and energy generation from metal fuel interaction under severe accident conditions.

These favorable characteristics suggest potentially more benign severe accidents for the Lungmen NPS than those analyzed in the SSAR.

The SSE for Lungmen is 0.4g, as opposed to the certified ABWR SSE of 0.3g. Although this design basis earthquake may result in larger seismic capacity for structures, systems, and components (SSCs), the PSAR seismic PRA did not take credit for such a potential advantage.

The PSAR seismic PRA indicate that the risk peaks at ground acceleration around 0.9g (See Figure AC.10-3). This is lower than the fuel seismic capacity (1.8g) and much lower than the RPV capacity (5.0 g). This indicates that the severity of core meltdown accidents will be primarily impacted by non-seimic fuel characteristics and by the initial and boundary plant conditions leading to core meltdown. The PSAR estimated the annual frequency of these initial and boundary conditions, or equivalently core damage categories, using the plant-specific seismic hazard curve. Since the RPV, RPV internals, and the fuel will not be significantly impacted by the seismic events, the fuel characteristics discussed above will determine the accident progression. Consequently, the SSAR Level 2 analysis has been judged as providing a conservative analysis for estimating the source term and radioactive release.

The seismic PRA reported in the PSAR is based on the Lungmen site seismic hazard curve. The SSCs seismic fragilities were either generic values or values based on an SSE of 0.3g. Only for those components where enhanced seismic design was found necessary to meet the Lungmen safety goals, upgraded seismic capacities were identified as design requirements. Table AC.3-10 contains the list of these components. Meeting the seismic capacity requirements of Table AC.3-10 will be verified when the design of these components is completed.

BOP design differences, e.g. the Lungmen capability for 110% steam bypass and electric load rejection, have been modeled in the PSAR Level 1 analysis.

As a general rule, key PRA assumptions, models, and data will be reviewed for applicability to the design as it evolves and for consistency with operating procedures as they are finalized. The validation of PRA assumptions and necessary updating of the PRA risk models will be performed in stages which are scheduled based on the Lungmen Project schedule for developing final system design descriptions, plant and equipment layouts, structural (including seismic) analysis, safety analysis, RPS hardware and software design, control room and simulator design and operating procedures,...etc. This stage-wise updating of the PRA models allows for timely feedback to the design process. It is conceivable that this PRA-design process interaction will involve iterations. The final risk model and analysis will be reported in the FSAR and will be consistent with the final design and operating procedures as described in the FSAR.

4. Section AB.6.11.3.3.2 reference to Section AJ.4.2 is related only to the available time for recovery before RPV breach. A typographical error resulted in missing the following sentence which should appear before the last sentence of the second paragraph of Section AJ.4-.2, page AJ.4-2. The sentence provides a lower bound estimate for the available time for in-vessel recovery. Section AJ.4.2 will be revised to include the following sentence.

   "Based on MAAP Code analysis reported in this attachment, it is expected that the in-vessel recovery would be possible for at least one hour from the initial loss of coolant injection."

5. The shutdown CET of Figure AG.8-2, page AG.8-13, is similar to the at-power CET for core damage category ID of Figure AB.6-8. Both CETs assign a value of 0.8 for the probability of RHR recovery if *either*

active injection *or* passive mitigation by passive flooder is successful. Please see the attached event trees. Table AB.6-5, page AB.6-38 will be revised to indicate that the probability of RHR recovery is 0.0 if *both* active injection *and* passive mitigation fail. Section AG.8.4.1.8, page AG.8-7 will also be revised to include discussion of the case for failure of both active injection and passive mitigation.

6. The MAAP3.0B -ABWR was developed for the analysis needed for certification of the Standard ABWR which was completed in 1994. Validity of the above analysis has been confirmed by the USNRC-sponsored analysis reported in Reference A-017.1. The reference includes analysis of five of the SSAR accident sequences using the MELCOR 1.8.2 Code and compares the results to those of the MAAP-ABWR. The main conclusions of the above comparison are:

   i) MAAP-ABWR and MELCOR produced similar time trends of key variables.

   ii) MELCOR generally predicts later times for core uncovery and slower core damage progression than MAAP-ABWR.

   iii) COPS disk rupture time and release fractions of radionuclides predicted by the two codes are comparable when debris quenching is included in MELCOR as it is in MAAP-ABWR.

References:

(1) L. N. Kmetyk, "MELCOR 1.8.2 Calculations of Selected Sequences for the ABWR," SAND94-0938, July 1994.

7. We agree. Validation of PRA key assumptions and their applicability to the as-built structures, systems, and components (SSCs) is scheduled based on the Lungmen Project schedule for developing the needed design and operating procedure information. Please see response to Item # 3 above for further discussion.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-018

PSAR Sections: Appendix A PRA

Question Date: June 15, 1998

PSAR Question:

The initiating events employed in Lungmen PRA study should be different from existing BWRs. Please clarify how the probabilities of those new events are determined.

Response:

PSAR Section A4 describes the approach for identifying a comprehensive set of initiating events for the Lungmen PRA. The identified initiating events cover different plant operating modes as well as external and internal challenges. The internal challenges are divided into those that have a plant-wide impact such as loss of off-site power, and those that are more localized. The fault tree of Figure A4-1presents the identified exhaustive set of initiating events which reflects operating experience and past PRA analyses. Section A4 discusses the rationale used to develop Lungmen-specific transients and LOCA events frequency from historic data and unique Lungmen design features such as the upgrade of low pressure piping to prevent ISLOCA. The following discussion provides further description of the approach for developing initiating events frequency and how it accounts for the differences between initiating events employed in the Lungmen PRA and existing BWRs. It is to be noted that full implementation of the following approach will be reported in the FSAR.

Differences between initiating events employed in the Lungmen PRA and existing BWRs may be divided into three categories: 1) Site-related events, e.g., seismic events which are site-specific, 2) Plant equipment-related events, e.g., LOCA from reactor recirculating water piping for BWRs but not for the Lungmen ABWR, and 3) Human action-related events, e.g., inadvertent control rod withdrawal.

1) Site-Related Events:

The Lungmen PRA of PSAR Appendix A is based on the seismic and typhoon hazard curves for the Yenliao site. These hazard curves provide the annual frequency of exceeding a given peak ground acceleration (seismic) or wind speed (typhoon). The qualitative screening of other site-related external events discussed in Section A.6 was also based on the Yenliao site.

Lungmen-specific annual frequency of loss of off-site power is used in the PSAR PRA. However, off-site recovery probability is based on US experience and will be updated for the Lungmen site in the FSAR.

The frequency of the above events is based on the Yenliao site historic data and standard frequency estimation techniques.

2) Plant Equipment -Related Events:

These events include internal fire and flooding. Lungmen-specific evaluation of the fire ignition sources, fire propagation, and annual frequency in risk significant plant locations is included in PSAR Attachment AD. Lungmen-specific internal flooding sources, water flow paths, and flooding annual frequency are included in Attachment AE.

Other equipment-related events include transients, LOCAs (inside and outside containment), ISLOCA, and single initiating events which may cause multiple failures.

The general approach for determining the frequency of the above Lungmen initiating events from BWR historic data and other applicable data involves five primary steps which may complement each other to ensure validity of the frequency values used:

a) Failure modes and effects analysis (FMEA) of new systems e.g. RIP, or system upgrades, e.g. low pressure piping upgrade to reduce the chance of ISLOCA. The purpose of the FMEA is to identify new initiating events and those traditional initiating events which may have been removed by design, supplemented by
b) Collecting available prototypical historic and test data of identified

initiating events. Applicability of the data is assessed by relating the initiating event to the failing system (or component), then comparing the design of the BWR system for which data has been collected to the corresponding Lungmen system to identify differences that could impact the initiating event frequency. The above assessment is supplemented by

c) Statistical data analysis and probabilistic analysis such as fault tree analysis to estimate initiating event frequency from prototypic historic or test data available for more basic events, supplemented by
d) Uncertainty and sensitivity risk analysis to assess the risk importance of the variability and accuracy of initiating event frequency values, supplemented by
e) IRA Program (PSAR Appendix B) activities to ensure that the performance of risk significant systems, structures, components (SSCs) and human actions remain under control and consistent with the frequency values used in the PRA during plant operation.

3) Human Action-Related Initiating Events:

The approach used to determine Lungmen-specific frequency of this category of initiating events is similar to that used above for the equipment-related events, where the BWR/Lungmen differences of interest include operating procedures, human/machine interface, and design provisions for mitigating human error.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-019

PSAR Sections: Appendix A PRA

Question Date: June 15, 1998

PSAR Question:

Please explain how the Goal values are determined in Table A-3 (A1-3) of Lungmen PSAR Appendix A. Please also provide comparisons of those values with similar, existing domestic and foreign plants.

Response:

PSAR Table A1-1 shows the references for the risk goal values used in Table A1-3. Specifically, the core damage frequency (CDF) and large early release frequency (LERF) goal values are from Reference 1 below. The individual and societal risk goal values are the USNRC Safety Goals specified in Reference 2.

The Lungmen CDF and LERF goals are similar to those defined in the Electric Power Research Institute (EPRI) Utility Requirements Document (URD) for US ALWRs. The USNRC does not specify CDF or LERF goals in Reference 2. Later analysis by the USNRC (Reference 3) of the Individual Plant Examination (IPE) PRAs for US nuclear power plants suggests that a CDF value of 1.0E-4 provides a sufficient condition for meeting the societal risk goal. The analysis also suggests that a LERF value of 1.0E-5 provides a sufficient condition for meeting the individual risk goal, where an early release refers to core damage accompanied by early containment failure and suppression pool bypass for BWRs. The USNRC intends to use the above CDF and LERF values in its risk-informed regulatory decisions (Reference 4). It is noted that:

1) The Lungmen CDF goal value of Table A1-3 (1.0E-5) is lower than the above USNRC CDF value by a factor of 10. This provides higher margin of investment protection and public safety, and is consistent with the US ALWR and future generation passive LWRs (e.g. SBWR) goals of enhancing the economics and safety of electric power

generation.

2) The USNRC definition of "large early release", which assumes no suppression pool scrubbing, leads to higher whole body site dose than the 0.25 Sv used for the definition of "large early release" for Lungmen. Moreover, the Lungmen LERF goal value of Table A1-3 (1.0E-6) is a factor of 10 lower than the above USNRC LERF goal value. The Lungmen conservative large early release definition and small LERF goal value are consistent with the US ALWR and future generation passive LWRs goals of enhancing the public safety of electric power generation.

Probabilistic safety goals like the USNRC individual and societal risk goals of Reference 2 have not been defined by regulatory authorities outside the USA. One of the high priority needs identified in Europe following the Chernobyl accident is the harmonization of the regulatory requirements between the European countries. The European PWR (EPR) is a joint German-French project being designed to meet common European Utility Requirements (EUR) which have been framed after the ALWR Requirements developed by EPRI in the USA. The EPR will meet the following EUR criteria (Reference 5):

- Global probability of core melt < 1.0E-5 per year, with the additional goal that the high pressure paths of core melt represent less than 10% of the global risk. This CDF goal is consistent with the Lungmen CDF goal of PSAR Table A1-3.
- Probability < 1.0E-6 to have a source term exceeding 100 Tbq of Cs.

The limit of 100 TBq of Cs stated above amounts to 2.0E-3 of the core inventory and has been included in the EUR criteria as a result of the Chernobyl accident. This is to be compared to the 1.0E-7 of the Cs inventory released via COPS in Lungmen severe accident sequences (See PSAR Appendix A, Table AJ.2-14, page AJ.2-33)

References

1. Taiwan Power Company Nuclear Island Bid Specification, 874-MS-001-1, Appendix A, Chapter 1, Appendix A: *Key Assumptions and Groundrules*, June 1995.

2. USNRC, *Safety Goals for the Operation of Nuclear Power Plants,*: 51 Federal Register 28044, August 4, 1986.

3. USNRC, Advisory Committee On Reactor Safeguards, Probabilistic Risk Assessment Subcommittee Meeting, Rockville, Maryland, USA, February 20, 1997

4. USNRC, "A*n Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to Current Licensing Basis - Draft*," Predecisional Regulatory Guide 1.174 (Draft Guide DG-1061), January 1, 1998.

5. Pierre Bacher, "European Utilities Objectives and Requirements for Future Nuclear Power Programmes," International Conf. on Design and Safety of Advanced Nuclear Power Plants, October 25 - 29, 1992, Toklyo, Japan, ANP'92, p1.2-1, October 1992.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-020

PSAR Sections: Appendix A PRA

Question Date: June 15, 1998

PSAR Question:

Even though it is more conservative to use risk bases of a foreign origin, but for comparison purposes it is suggested that domestic data be also employed to truly evaluate safety of design.

Response:

We agree that applicable domestic (i.e. Taiwanese) data should be employed in the Lungmen PRA. One of the primary objectives of the Lungmen PRA is to provide realistic plant-specific risk estimates of the as-built plant design and operation. This will be accomplished in two broad steps:

1) Use of prototypical data, including applicable domestic equipment and human error data, in the FSAR PRA.

2) Use of the IRA Program (PSAR Appendix B) to ensure that the performance of risk-significant systems, structures, components, and human actions are consistent with the probability and unavailability values used in FSAR PRA.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-021

PSAR Sections: Appendix A PRA

Question Date: June 15, 1998

PSAR Question:

Historic experiences showed that there were tsunami occurrences along the northern Taiwan coast. Please clarify whether the risk assessment has included seawater retreat that could possibly cause loss of coolant or even jeopardize core safety.

Response:

The Lungmen PRA included qualitative screening analysis of tsunami events to show that these events present insignificant risk. The analysis accounted for drawdown of the seawater that could lead to loss of the heat sink, as well as wave run-up that could cause flooding. The Lungmen ultimate heat sink (UHS) design meets Regulatory Guide 1.27, which requires the UHS functions to be assured during and following the most severe natural phenomena postulated for the site. As stated in PSAR Section 9.2.5.1 (3), page 9.2-5, the UHS is sized such that sufficient cooling water is provided during maximum tsunami drawdown. Please see PSAR Section 2.4.6 for the probable maximum tsunami flooding, Section 2.4.11 for low water considerations, and Section 9.2.5 for the ultimate heat sink design.

The Tsunami discussion in PSAR Appendix A, page A6-12, will be amended to include discussion of the risk from seawater drawdown caused by tsunami. This discussion will be included in the PSAR Amendment.

ROCAEC Review Comment:

Please provide the proposed PSAR changes for Tsunami discussion as committed in the above response.

Further Clarification :

Proposed changes to PSAR Page A6-12 are as follows:

Tsunami

Tsunami waves present two types of potential hazard: flooding that may be caused by wave run-up, and ultimate heat sink (UHS) loss that may be caused by seawater drawdown. Review of the Lungmen NPS design indicates that these hazards have been removed by design and need not be considered as initiating events for detailed PRA evaluation.

The Yen-Liao site is located above the worst case flood for the region. Section 3.1.4.5 of Chapter 3 of the TPC Bid Specification specifies that grade-level for the site will be 12 m above mean sea level (MSL). The probable maximum tsunami for the Yen-Liao site is developed in Subsection 2.4.6. As indicated in Subsection 2.4.6.5, the probable maximum tsunami water run-up level is 8.57 m MSL. Therefore, the maximum tsunami-induced flood level will be 3.43 m below plant grade.

Subsection 2.4.6.5 also shows a tsunami minimum drawdown water level of -8.68 m MSL. The RBSW intake is designed to ensure UHS availability under this condition. Specifically, the Lungmen UHS design meets Regulatory Guide 1.27, which requires the UHS functions to be assured during and following the most severe natural phenomena postulated for the site. As stated in Subsection 9.2.15.3, the RBSW pumps take suction from a submerged pond to guarantee service water during a maximum tsunami drawdown period. The UHS pond is sized such that sufficient cooling water is provided during maximum tsunami drawdown (Subsection 9.2.5.1 (3)). Therefore, the minimum drawdown seawater level will not disrupt the UHS availability.

The above preventive design features provided the basis for screening tsunami events from further PRA consideration.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: B-001 and B-002

PSAR Sections: Appendix B

Question Date: February 9, 1998

PSAR Question:

Track B-001

1. A method similar to PRA was said will be adopted for the Reactor Scram in the PSAR. It is requested that a detailed description of the method of analysis and its contents be provided.

2. Finding the failure modes of SSC based on historical records will not always encompass all the failure modes of the same equipment. Please re-evaluate if anything is overlooked.

3. Please explain in detail how the "preventive maintenance" (PM) is decided for SSC. Does it include considerations of supplier's design information, past maintenance experience or FMEA analysis results that show the correctness of the PM program ?

Track B-002 (Related to Item 1 of B-001 above)

B.3.1.1.2 and B.3.1.2.1 Unplanned Outage Analysis indicated that S&W will develop the model regarding the BOP non-safety equipment which will affect the reactor trips and establish the database for related parameter. Based on the output of this model analysis, will provide the key data for critical components for which may affect the reactor trips. ROC-AEC ask TPC to provide the analysis model and methodology which were developed by S&W, in order to verify whether the analysis model meet the requirements.

Response:

1. **(Includes B-002)** Lungmen Integrated Reliability Analysis (IRA) Program for unplanned outage analysis will include the reactor trip

frequency and forced outage as described in Appendix B. GE is responsible for the integration of Nuclear Island and Balance of Plant portions of the Lungmen integrated reliability analysis. The program is being developed and the methodology for IRA unplanned outage analysis will be described in the Final Safety Analysis Report. The description of the preliminary methodology will be incorporated into PSAR Appendix B as a new Attachment BE, Methodology For Unplanned Outage Analysis. Therefore, GE will add, "(See Attachment BE)" after sentences ".....that used in the PRA." in Sections B.3.1.1.2 and B.3.1.2.1. Attachment BE will be included in the PSAR as follows:

Attachment BE: Methodology For Unplanned Outage Analysis

The IRA evaluation will use proven fault tree analysis techniques, consistent with PRA models, to develop fault trees for each system and for the integrated plant to assure that reactor trip frequency and forced outage performance standard will be met. Since unplanned outage analysis will be part of the results from Lungmen IRA evaluation, the process for Lungmen IRA evaluation is provided. The major steps and summary of Lungmen IRA analysis activities are described in the following steps:

Step 1: Definition of plant and system success criteria

The first step of the system analysis is to determine the system impact to support the plant operation. Some systems can be in a test or maintenance alignment and still support full power operation, whereas other system may need to operate in a normal alignment to prevent plant tripping off line. The following assessment will define the plant and system success criteria. The typical items to be reviewed are:

- *identification of system dependencies*
- *determination of system success criteria to support plant operation*
- *capability to maintain equipment with plant in power operation*
- *limiting conditions of operation in Technical Specifications*

Step 2: Plant level and system level fault tree construction

The goal of this step is to find combinations of events that impact power operation which may result in a sudden plant trip or manual shutdown. For the plant level fault tree the top event for the plant is, "Plant Unable to Generate Electrical Power." The top logic of this fault tree accounts for all ways that may cause interruption to power operation in terms of hardware states and human actions. The following system alignments or conditions which will impact plant operations are:

- *events that cause a need to effect an orderly plant shutdown to correct a problem*
- *events that would cause a sudden reactor trip*

The plant level fault tree will develop the logic until system interfaces are identified. These system interfaces will be used to define top events for system level fault trees. Once the system level fault trees are developed, they will be integrated into a single large linked fault tree to perform the integrated plant analysis function.

Step 3: Minimal cutset determination

The implementation of this step is the same as the standard fault tree solution with one exception: special flags must be used to prevent Boolean reduction of certain cutsets that are not important for estimating the fault tree top event probability but are extremely important to support implementation of this methodology.

Once the power reduction and shutdown mode flags have been placed on the tree, Boolean reduction of the model can proceed with a standard fault tree analysis.

Step 4: Definition of scenarios for each cutset

The purpose of this step is to define a set of scenarios that are derived from the fault tree minimal cutsets. For each cutset, there are one or more scenarios that meet the conditions for the cutset to occur. Each scenario has an initiating event and a set of plant conditions which are necessary to produce the cutset conditions. The scenarios are developed from each cutset using a set of simple rules summarized as follows.

- *For each basic event corresponding to normally operating equipment, the events include the equipment failure to continue operating or the equipment is taken out of service for testing or maintenance.*

- *For each scenario initiating event identified for a given cutset, there may be one or more individual scenarios that would meet the condition for a cutset to occur. For example it there is a cutset {A,B} the typical scenarios for this cutset might include:*

  - Failure of A with B out of service for maintenance
  - Failure of A and B fails while A is being repaired

- Failure of A and B fails to start on demand

- *For basic events associated with either standby or operating equipment subject to limiting conditions of operation (LCO), there may be manual plant shutdown scenarios for an LCO and remaining out of service that exceeds the LCO requirement. There may also be anticipatory manual shutdowns for basic events which degrade the plant ability to continue to operate, but which do not cause an immediate trip.*
- *For critical test and maintenance alignments, there may be other scenarios for operator induced trips.*

Step 5: Modeling and quantification of scenario frequencies

The system level models may include, as appropriate:

- *failure rates and repair times for scenario initiating events and basic events*
- *test and maintenance unavailability models*
- *models for probability that an equipment is out of service longer than LCO requirement*
- *human reliability models*
- *common cause failure models*

Step 6: Characterization of scenario impacts

This task is to characterize the consequences of each scenario which will cause plant trip or manual plant shutdown. The duration of any plant outage is assessed in terms of time when generator is off line for the purpose of determining the plant availability.

Step 7 : Integrated plant quantification for reliability and availability

Utilizing the information in the previous steps, the analysis tool will provide an integrated plant assessment in terms of the following plant level performance measures which includes frequency of plant trips:

- *plant availability factor*
- *forced outage rate*
- *frequency of plant trips*
- *expected hours of forced outage/year*

2. The determination of dominant failure mode of risk-significant SSCs will include historical information, analytical models and existing requirements. The process for determining dominant failure modes of risk-significant SSCs is described in Figure B-3 of PSAR. Use of the failure history to determine failure mode is one of the assessment paths as describe below.

   Failure history to define failure modes, as described as assessment path A in Figure B-4 of PSAR, is used for less complex equipment when failure history data is available. Since a reasonable long failure history is necessary for most components to determine the dominant failure modes from failure and repair data, it may be useful to combine components into categories that allow pooling, or mixing of the failure histories from several components. The first step in this option is to develop the analysis boundary in terms of categories of equipment whose repair and failure data would be pooled. The next step of this option is to construct the list of failure modes found in the failure data. This could be accomplished in terms of piecepart failure using piecepart failure cause data. If the piecepart failure data is not available, the list should be constructed by major piecepart failure. The occurrence frequency of each category is then computed, and the categories are ranked by occurrence frequency, with the most frequently occurring piecepart failures indicated as the prime candidates for inclusion as the dominant failure modes.

   The assessment path B, analytical assessment to define failure modes, as described in Figure B-5 of PSAR is used for complex equipment, or when failure history data are not available. In this option, a qualitative analytical tool such as fault tree, Failure Mode and Effect Analysis (FMEA), or reliability block diagram is used to identify pieceparts of risk-critical components as shown in the large box of Figure B-5.

3. In addition to the assessment path A and B to define the dominant failure mode list as described in Question 2 response, the process to review existing maintenance related activities and requirements are described in Figure B-6 of the PSAR. The ASME Section XI requirements, vendor recommendations, Technical Specification

requirements, environment requirements and other regulatory-mandated requirements will be listed to the associated risk-significant SSCs. This explicit set of steps could serve as a starting point for the assessment of maintenance needs for the component. These recommended maintenance activities are further reviewed to identify failure modes affected and frequency of maintenance, and identify if there are any failure modes that are not maintained. The process as defined on Figure B-6 will identify the maintenance strategies to defend against the dominant failure mode of risk-significant SSCs.

The process of determining the risk-significant SSCs preventive maintenance program considers the vendor recommendations, failure mode and effect analysis or failure history. The performance monitoring of the risk-significant SSCs during plant operation will provide the feedback of the maintenance program effectiveness and, if needed, the maintenance activities will be revised to defend any newly identified dominant failure mode.

The PSAR will be revised to add a new Attachment BE as described in the response of item 1 above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: B-002

PSAR Sections: Appendix B

Question Date: March 23, 1998

PSAR Question:

B.3.1.1.2 and B.2.1.2.1 Unplanned Outage Analysis indicated that S&W will develop the model regarding the BOP non-safety equipment which will affect the reactor trips and establish the database for related parameter(s). Based on the output of this model analysis, [it] will provide the key data for critical components which may affect the reactor trips. ROC-AEC ask TPC to provide the analysis model and methodology which were developed by S&W, in order to verify the analysis model meet(s) the requirements.

PSAR Response:

The following description of the IRA evaluation applies to the BOP system models under development by S&W that will include the contributors to unplanned plant trips and forced outages as stated in Appendix B, Lungmen PSAR, paragraphs B3.1.1.2, and B3.1.2.1.

The IRA evaluation will use proven fault tree analysis techniques, consistent with PRA models, to develop fault trees for each system and for the integrated plant to assure that reactor trip frequency performance standard will be met. Since reactor scram frequency will be part of the results from Lungmen IRA evaluation, the process for Lungmen IRA evaluation is provided. The major steps and summary of Lungmen IRA analysis activities are described in the following steps:

Step 1: Definition of plant and system success criteria

The first step of the system analysis is to determine the system impact to support the plant operation. Some systems can be in a test or maintenance alignment and still support full power operation, whereas other system may need to operate in a normal alignment to prevent plant tripping off line. The following assessment will define the plant and system success criteria.

- identification of system dependencies
- determination of system success criteria to support plant operation
- capability to maintain equipment with plant in power operation
- limited conditions for operation in technical specifications

Step 2: Plant level and system level fault tree construction

The goal of this step is to find combinations of events that could cause perturbation to power operation which may result in a sudden plant trip or manual shutdown. For the plant level fault tree the top event for the plant is "Plant Unable to Generate Power". The top logic of this fault tree accounts for all ways that may cause interruption to power operation in terms of hardware states and human actions. The following system alignments or conditions which will impact plant operations are:

- events that cause a need to effect an orderly plant shutdown to correct a problem
- events that would cause a sudden reactor trip

The plant level fault tree will develop the logic until system interfaces are identified. These system interfaces will be used to define top events for system level fault trees. Once the system level fault trees are developed, they will be integrated into a single large linked fault tree to perform the integrated plant analysis function.

Step 3: Minimal cutset determination

The implementation of this step is the same as the standard fault tree solution with one exception: special flags must be used to prevent Boolean reduction of certain cutsets that are not important for estimating the fault tree top event probability but are extremely important to support implementation of this methodology.
Once the power reduction and shutdown mode flags have been placed on the tree, Boolean reduction of the model can be proceeded with a standard fault tree analysis.

Step 4: Definition of scenarios for each cutset

The purpose of this step is to define a set of scenarios that are derived from the fault tree minimal cutsets. For each cutset, there are one or more scenarios that meet the conditions for the cutset to occur. Each scenario has an initiating event and a set of plant conditions which are necessary to produce the cutset conditions. The scenarios are developed from each cutset using a set of simple rules summarized as follows.

- For each basic event corresponding to normally operating equipment, the events include the equipment failure to continue operating or the equipment is taken out of service for testing or maintenance.
- For each scenario initiating event identified for a given cutset, there may be one or more individual scenarios that would meet the condition for a cutset to occur. For example it there is a cutset {A,B} the typical scenario for this cutset might include:
  - Failure of A with B out of service for maintenance
  - Failure of A and B fails while A is being repaired
  - Failure of A and B fails to start on demand
- For basic events associated with either standby or operating equipment subject to limiting conditions of operation (LCO), there may be manual plant shutdown scenarios for an LCO and remaining out of service that exceeds the LCO requirement.
- For critical test and maintenance alignments, there are scenarios for operator induced trips

Step 5: Modeling and quantification of scenario frequencies

The standard models include, as appropriate:

- failure rates and repair times for scenario initiating events and basic events
- test and maintenance unavailability models
- models for probability that an equipment is out of service longer than LCO requirement
- human reliability models
- common cause failure models

Step 6: Characterization of scenario impacts

This task is to characterize the consequences of each scenario which will cause plant trip or manual plant shutdown. The duration of any plant outage is assessed in terms of time when generator is off line for the purpose of determining the plant availability.

Step 7 : Integrated plant quantification for reliability and availability

Utilizing the information in the previous steps, the analysis tool will provide an integrated plant assessment in terms of the following plant level performance measures:

- plant availability factor
- forced outage rate
- frequency of plant trips
- expected hours of forced outage/year

The integrated plant model, for which GE is responsible, combines the analyses for both BOP systems, and reactor plant systems. When completed, it will identify critical components which may affect the frequency of plant trips and forced outage rate.

No changes will be made to the PSAR as a result of the response to the question.

ROCAEC Review Comment:

Responses to item one of PSAR question B-001 and B-002 are identical. However, GE committed to incorporate the changes identified in B-001 into PSAR Amendment but the response to B-002 "REPLY TO TPC REVIEW COMMENTS" stated that "No changes will be made to the PSAR as a result of this question". Please clarify and also include the 7 steps into PSAR from the original GE response.

Further Clarification:

The question and response are combined with Track No, B-001 based on ROCAEC comment.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track　Number)：　　P-C-001

問題章節(PSAR Section)：　　附錄C

初提日期(Question Date)：　　1997.12.15

問題內容(PSAR Question)：

PSAR的撰寫應明確說明在核四廠建廠過程中，台電公司欲（或必須）承諾的事項為主。本附錄C中所有的資料均屬一般性的描述，並未針對核四部分做出說明，宜大幅度的改寫。有關具體審查意見，將分別於相關章節中提出。

問題答覆(Responses)：

核四PSAR撰寫之法規依據，包括有：行政院〝原子能法規〞、〝核子事故緊急應變計畫〞、〝核四環評報告〞、台電〝緊急計畫準則〞以及參考美國核能相關法規等，且均係以台電公司承諾之事項為主。

原能會審查意見(ROCAEC Review Comments)：

台電公司應再行提出針對核四部分的說明，並在PASR報告中所承諾之事項，以表列或條列方式明確敘述。

台電澄清說明(Further Clarification)：

已遵照指示，將台電在核四廠建廠過程中具體承諾辦理之事項列表如附。

此外亦根據各項體審查意見改寫並重新編排相關章節內容以反映核四廠之狀況。對各項審查本公司初次答覆之意見之澄清說明，以新編寫之附錄C為準。

原能會審查(二)意見(ROCAEC Review Comments)：

台電公司承諾事項中提到多項將協調全委會作業執行室的工作，是否已與全委會相關人員協商過？時程上是否可以相互配合？。

台電澄清說明(Further Clarification)：

核能四廠緊急計畫涉及廠外部份事項，本公司已於87年9月7日與全委會作業執行室開會協調，將由全委會作業執行室主辦，台電協辦；其中主要工作為於核能四廠核燃料裝填前完成核四廠廠外緊急計畫作業程序書，並完成下列各主要項目：

1. 建立核四廠廠外民眾「核子事故預警系統」。
2. 依據核四廠緊急計畫區內人口分佈、殘障居民、交通流量、道路狀況、遊樂設施與遊客動態等資料之調查統計與分析結果，完成評估現有道路作為疏散用之妥適性，及建立包括遊客在內之有效疏散路網分析模式。
3. 在FSAR中明確說明民眾宣導資料(包括如定期刊物等)的規劃、印製與發送。
4. 明確標出可能使用的集結點。
5. 明確劃定輻射偵測路線及偵測點。
6. 明確規畫近指中心設置地點。若規劃地點位於緊急計畫區半徑範圍內，將另覓適當地點做為後備近指中心。

註1： 上述第2項工作中之核四廠緊急計畫區人口分佈、殘障居民、交通流量、道路狀況、遊樂設施與遊客動態等資料之調查統計與分析將於核燃料裝填前二年完成，至於評估現有道路作為疏散用之妥適性，及建立包括遊客在內之有效疏散路網分析模式則將於核燃料裝填前半年完成。

註2： 上述第6項之近指中心已於會後選定設於放射實驗室核能四廠工作隊旁之一單獨建築物內。

原能會審查(三)意見(ROCAEC Review Comments)：

C.1.4.3.2各項承諾應明確設定完成之時間，例如商業運轉前，或繳交FSAR時。

台電澄清說明(Further Clarification)：

已遵照辦理，修正如C.1.4.3.2。

原能會審查(四)意見(ROCAEC Review Comments)：

C.1.4.3.2 須對構想中之"核子事故預警系統"提出簡略之說明。

台電澄清說明(Further Clarification)：

已遵照辦理，修正如C.1.4.3.2。

原能會審查(五)意見(ROCAEC Review Comments)：

建議將"承諾表"正式放入附錄中，將來更容易追蹤管理。

台電澄清說明(Further Clarification)：

"承諾表"中之承諾事項已於附錄C中各相關章節明確承諾辦理。

P-C-001之附件

龍門PSAR附錄C台電承諾事項一覽表

編號	章節	頁次	台電公司承諾事項	＊完成期限	備註(相關審查問題編號)
001	EIA (附錄C第6.2節)	#24 (C.6-6)	完成核四廠緊急應變計畫之編訂，並送呈原能會審核。	A	------
002	EIA (附錄C第6.2節)	#24 (C.6-6)	編妥核四廠緊急計畫實施程序，經本公司核安會審核後送原能會備查。	B	P-C-051
003	EIA (附錄C第6.2節)	#24 (C.6-6)	編妥核四廠全廠(廠內)緊急計畫演習方案，經本公司核安會審核後送原能會備查。	B	P-C-051
004	EIA (附錄C第6.2節)	#24 (C.6-6)	執行核四廠第一次全廠(廠內)緊急計畫演習。	B	P-C-051
005	EIA (附錄C第6.2節)	#24 (C.6-6)	建立核四廠廠外民眾「核子事故預警系統」。	C	------
006	EIA (附錄C第6.2節)	#25 (C.6-6)	依據核四廠緊急計畫區內人口分佈、殘障居民、交通流量、道路狀況、遊樂設施與遊客動態等資料之調查統計與分析結果，協調全委會作業執行室完成評估現有道路作為疏散用之妥適性，及建立包括遊客在內之有效疏散路網分析模式。	C	P-C-039 P-C-040
007	附錄C 第3.2節	(C.3-26)	具體提出簽約醫院(例如三總)對台電核四廠提供緊急醫療照護與輻傷救護措施之承諾書(或合約等)，以及簽約醫院與核四廠之相對地理位置圖。	A	P-C-005

龍門PSAR附錄C台電承諾事項一覽表(續頁)

編號	章節	頁次	台電公司承諾事項	＊完成期限	備註(相關審查問題編號)
008	附錄C第5.13節 EIA	(C.5-156)	台電過去委託簽約醫院成立之「輻射傷害防治中心應落實在核四廠的緊急計畫中。	A	P-C-005
009	附錄C第4.1節	(C.4-1)	因應未來科技之進步與發展，增設與提升緊急通訊設施之硬體裝備。	B	P-C-007
010	附錄C	------	將來隨著科技之進步，隨時密切注意國外最新發展趨勢，研發事故時整合機組狀況、劑量分佈、氣象條件、交通狀況、預估劑量等資訊且適合核四廠之電腦軟體以提供全委會廠外民眾劑量及防護行動建議。	A	P-C-008
011	附錄C	------	將協調全委會作業執行室後，在FSAR中明確說明民眾宣導資料(包括如定期刊物等)的規劃、印製與發送。	A	P-C-009
012	附錄C 第5.11.2節	(C.5-147)	將協調全委會作業執行室後，明確標出可能使用的集結點。	B	P-C-041
013	附錄C 第5.6.5節	(C.5-87)	明確劃定輻射偵測路線及偵測點。	B	P-C-044
014	附錄C 第4.2節	(C.4-25)	設置近指中心。	B	P-C-050

＊註1： A：FSAR完成提報階段。
B：第一部機組初始燃料裝填前。
C：第一部機組運轉前。

註2； 編號010之承諾係於FSAR完成提報階段提出初步規劃評估報告。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： P-C-002

問題章節(PSAR Section)： 附錄C

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

PSAR內容若涉及其他單位（如：全委會作業執行室）時，台電公司應主動循正式程序協調定案後提供具體作法，而非逕行抄錄舊有資料提報。

問題答覆(Responses)：

PSAR內容廠外緊急計畫部分係依據核子事故緊急應變計畫(行政院83.3修訂頒布)、全委會廠外緊急計畫作業程序書，並參考美國聯邦法規、核管會法規指引等予以編寫，且在必要時，本公司將會主動協調廠外其他單位(如全委會作業執行室)定案後提供具體作法。

原能會審查意見(ROCAEC Review Comments)：

本意見之提出，是希望台電公司能在PSAR報告中作出明確之承諾，如：本公司承諾在進行××××事物之協調時，將主動依××××之規定與××××等單位協調定案後，於FSAR中提出××××等之具體方案。

台電澄清說明(Further Clarification)：

已遵照補充承諾如本附錄C第1.4.3節〝核能四廠運轉前緊急應變之準備作業〞。至於有關緊急計畫廠外部份有關事項，本公司已於87年9月7日與全委會作業執行室開會協調，對附錄C所述相關內容亦討論獲共識。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： P-C-003

問題章節(PSAR Section)： 附錄C

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

PSAR是一套必須公開陳列的電廠相關文件，且緊急計畫與電廠附近的民眾息息相關，所以在緊急計畫的編寫與審查上，絕對不能草率。但是從目前的附錄C內容來看，緊急計畫的規劃，並未獲得台電公司應有的重視。

問題答覆(Responses)：

(1) 附錄C緊急計畫係本公司依據國內外許多相關法規之規定，並協調廠內外有關單位後，以相當審慎的態度編寫而成。

(2) 附錄C內容(草案)，先前經送請奇異與石偉公司審查後參考修訂，過程完全符合審查程序之要求。

原能會審查意見(ROCAEC Review Comments)：

此問題的答復非常不妥當。附錄C雖送請奇異與石偉公司審查，但其審查並未考量我國緊急應變計畫之特性，似無實質意義。附錄C大部分內容仍未能符合要求，台電公司應重新認真編寫並加強內部審查作業之品質。

台電澄清說明(Further Clarification)：

(1)本附錄在PSAR階段原即著重在制度面規劃性之說明。目前本公司已遵照各審查意見再就各章節內容做進一步之說明，詳如各題答覆內容。

(2)附錄C緊急計畫係本公司依據國內外許多相關法規之規定，並協調廠內外有關單位後編寫而成。

(3)已遵照審查委員意見重新認真編寫並加強內部審查作業之品質。

編號(Track Number)： P-C-004

問題章節(PSAR Section)： 附錄C

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

報告中似乎提到或陳述太多原能會或全委會的工作與職掌，但台電公司本身應規劃或承諾的工作反而不甚清楚。例如：C-75～C-98頁所提之近指中心似乎不應屬台電之職掌。台電公司應站在身為核反應器設施經營者立場，提出如何協助廠外組織進行輻射偵測與成立近指中心，以及可能做到的承諾。

問題答覆(Responses)：

台電緊執會所屬環境偵測組在近指中心成立前執行並預作準備工作，俟近指中心成立後即正式併入該中心輻射偵測隊編組中執行緊急應變任務，在台電緊急計畫準則及該組作業程序書中均有明文規定。

原能會審查意見(ROCAEC Review Comments)：

台電公司對於問題之答復並未切題。問題的內容是要求台電在PSAR中提出如何協助其他組織執行緊急應變任務的承諾。例如：台電公司可以對輻射偵測事宜提出如下之承諾---本公司承諾提供廠外其他組織，如：地方政府及民間環保監督團體，必要之協助進行平時與緊急應變時相關之輻射偵測事宜。

台電澄清說明(Further Clarification)：

已遵照指示增加台電協助成立近指中心之說明於附錄C第4.2節〝近廠指揮協調中心設置說明〞、第5.6節〝緊急計畫區內輻射偵測隊派遣說明〞、及第5.7節〝廠內技術支援中心與近廠指揮協調中心在訊息評估、輻防建議及民眾訊息傳遞之任務說明〞各節內。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： P-C-005

問題章節(PSAR Section)： 附錄C

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

報告中提到數項台電公司過去委託學術單位執行的計畫，如：C-65頁提到TEVACS，C-128頁提到成立輻射傷害防治中心。台電公司是否表明會將這些已完成之成果落實在核四廠的緊急計畫中？但如何落實似乎沒有詳述。

問題答覆(Responses)：

(1) 擬修訂附錄C Ⅲ.10〝廠外緊急醫療照護與輻傷救護措施〞內容並補充具體說明。

(2) 附錄 C Ⅲ.11〝簽約醫院接受治療傷患能力評估〞中提到有關設備、人力及能量等具體資料，可供參考。

原能會審查意見(ROCAEC Review Comments)：

台電公司亦可承諾將於FSAR中具體提出如：

(1)簽約醫院對核四支援的承諾書。

(2)醫院及電廠相關位置圖。

(3)輻傷運送計畫等。

並提出具體可行落實過去委託學術單位執行計畫之方案。

台電澄清說明(Further Clarification)：

(1) 已修訂附錄C第5.13節〝廠外緊急醫療照護與輻傷救護措施〞內容並補充具體說明。

(2) 請參閱本附錄C第5.14〝簽約醫院接受治療傷患能力評估〞。

(3) 本公司承諾在FSAR中明確說明以下事項：

將來簽約的醫院。

簽約醫院對核四提供緊急醫療照護與輻傷救護措施之承諾書。

簽約醫院距離核四之相關地理位置圖。

(4) 有關事故時之核四廠輻射傷患緊急送醫計畫，包括廠內緊急醫療救護除污與傷患送醫措施等說明已補充如本附錄C第5.12節〝廠內緊急醫療救護除污與傷患送醫措施〞。

(5) 以上本公司承諾事項如審查問題編號P-C-001附表「龍門PSAR附錄C台電承諾事項一覽表」，#007)。

編號(Track Number)： P-C-006

問題章節(PSAR Section)： 附錄C

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

報告中未提到如何建立因應電廠緊急事故時廠內及廠界之輻射偵測計畫及監測網。

問題答覆(Responses)：

(1) 有關核四廠廠內環境輻射偵測在龍門PSAR第12章"Radiation Protection"中已有深入且詳盡之說明，故建議無需在附錄 C "緊急計畫"中重複敘述。

(2) 有關核四廠廠外環境輻射監測，本公司將依 貴會頒佈之「環境輻射偵測規範」規劃執行，並將於龍門FSAR第11.6節"Offsite Radiological Monitoring Program"詳細說明，故建議無需在附錄C〝緊急計畫〞中重複敘述。

原能會審查意見(ROCAEC Review Comments)：

PSAR 12章及11.6節乃針對一般正常運轉狀況進行說明，緊急應變時狀況有所不同，且部分非緊急應變人員可能撤離，故應有區別說明。

台電澄清說明(Further Clarification)：

(1) 已補充說明有關如何建立因應電廠緊急事故時廠內及廠界之輻射監測計畫及監測如附錄C第5.8節〝廠內輻射防護措施〞。

(2) 上述未盡事宜，本公司將協調全委會後在FSAR中明確說明核四廠緊急輻射監測計畫及劃定輻射偵測路線及偵測點，並承諾如審查問題編號P-C-001附表「龍門PSAR附錄C台電承諾事項一覽表」，#013)。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： P-C-007

問題章節(PSAR Section)： 附錄C

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

隨著科技的進步，各項通訊、電腦設備早已日新月異，但本報告顯然仍抄襲舊有的資料，未作考慮更新。例如：無線電話（大哥大）、電視傳真、網路系統等均未列入（C-105頁）。

問題答覆(Responses)：

茲將本章節中有關OSC通訊設施最低要求之相關規定摘錄如下：

----包括電話、傳真設備各一具以上……，並無不妥。至於新式科技產品如大哥大、電腦傳真、網路系統等，依據本公司「核能電廠緊急計畫準則」並無相關之規定，故不擬硬性規定。

原能會審查意見(ROCAEC Review Comments)：

現有運轉中之核能電廠多已建置或評估設置多媒體TV系統、大哥大、無線電對講機等通訊設施，核能四廠更應規劃設置較先進、完善的通訊設施。

台電澄清說明(Further Clarification)：

(1) 已遵照增加規劃設置更先進的通訊設施如本附錄C第5.2節〝通知各緊急應變組織及民眾所用方法與時間〞及圖C.5.2-1。

(2) 本公司將會依未來科技之進步與發展，增設並提升緊急通訊設施之硬體裝備，以進一步強化通訊能力，並承諾如審查問題編號P-C-001附表「龍門PSAR附錄C台電承諾事項一覽表」，#009)。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： P-C-008

問題章節(PSAR Section)： 附錄C

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

台電公司應說明事故時如何整合機組狀況、劑量分布、氣象條件、交通狀況、預估劑量等資訊，以提出廠外民眾劑量及防護行動建議。

問題答覆(Responses)：

(1) 事故時台電公司緊執會之劑量評估組將依氣象資料、輻射外釋狀況據以預估廠外民眾劑量，並按時蒐集由環境偵測組傳來之廠外環境偵測結果，比較及修正預估之廠外民眾劑量值，其作業流程如附圖所示。

(2) 上述之結果由劑量評估組組長向緊執會委員會報告，再由委員會依據運轉支援組之機組運轉狀況報告與事故評估組之機組安全狀況報告，由委員會所有委員共同加以綜合評估研判，如有突發之特殊交通狀況等一併納入考慮，以作出最佳之決策提供廠外民眾輻射防護行動建議。

(3) 因本公司緊急計畫之權責單位為緊執會，基於組織型態的考量其組統架構係採功能委員制，下設九個工作組，各司不同之專門功能，由各工作組向委員會報告並提供分析結果與建議後，由全體委員集思廣益據以研判並作出最佳決策。

(4) 以本公司緊急應變組織之結構而言，無法也無需藉一次整合機組狀況、劑量分佈氣象條件、交通狀況、預估劑量等資訊之電腦軟體，以提出廠外民眾防護行動建議，且據悉目前世界上核能先進國家似尚無此種作法可供參考。

原能會審查意見(ROCAEC Review Comments)：

本問題強調的是台電公司需對民眾防護措施建議之決策過程有所描述

說明；另台電公司並未針對本問題提出明確答復是否有誠意去整合與提出具體可行之行動建議。

台電澄清說明(Further Clarification)：

(1) 事故時台電公司緊執會之劑量評估組將依氣象資料、輻射外釋狀況據以預估廠外民眾劑量，並按時蒐集由環境偵測組傳來之廠外環境偵測結果，比較及修正預估之廠外民眾劑量值，其作業流程如附圖所示。

(2) 由緊執會執行秘書協調劑量評估組上述之預估結果，連同運轉支援組之機組運轉狀況報告及事故評估組之機組安全狀況報告提出綜合報告，由主任委員主持，所有委員共同加以審查討論後，提出最佳之廠外民眾輻射防護行動建議。

(3) 本公司目前已委託專家學者針對核四廠之設備安全建立一電腦化快速評估系統。

(4) 本公司承諾，將來隨著科技之進步，隨時密切觀察國外最新相關整合評估軟體發展之趨勢，積極研發一套可以整合機組狀況、劑量分佈、氣象條件、交通狀況、預估劑量等資訊且適合核四廠之電腦軟體，以提供緊急事故時全委會最佳之決策工具。

(5) 以上本公司承諾事項如審查問題編號P-C-001附表「龍門PSAR附錄C台電承諾事項一覽表」，#010)。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： P-C-009

問題章節(PSAR Section)： 附錄C

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

台電公司應說明有關民眾宣導與教育的規劃及預定執行的方式。

問題答覆(Responses)：

1. 台電公司將針對核能四廠廠區附近民眾定期發送「鄉土情」月刊，闡述核能發電與安全之相關事項，以灌輸民眾清楚而正確的核能發電安全方面知識。
2. 台電公司將每年定期利用寒假期間對核能四廠附近學校(主要以國中生為主)分批舉辦核能發電與安全之訓練課程，並有一系列之相關活動，以增進學生對核能發電的了解。

原能會審查意見(ROCAEC Review Comments)：

台電公司應就民眾教育及溝通事宜，提出更具體的說明與規劃。定期發送之刊物不宜限定「鄉土情」，台電公司出版之各類核電知識、核安資訊等刊物均可納入；訓練課程則應針對不同對象擬訂適當之課程內容、期程、頻次等，以求落實有效。

台電澄清說明(Further Clarification)：

(1) 有關民眾宣導資料方面：

台電公司已針對核能四廠廠區附近民眾定期發送適當之核能宣導刊物，內容為闡述核能發電與安全等相關事項，以灌輸民眾清楚而正確之一般核能發電安全方面知識。

本公司將協調全委會作業執行室後，在FSAR中明確說明印製發送附有如緊急疏散道路、民眾預警系統等有關緊急計畫宣導資料(如農民曆等)。

(2) 有關教育規劃及執行方式方面：

台電公司將持續目前做法，每年定期利用寒假期間對核能四廠附近學校(主要以國中生為主)分批舉辦一般核能發電與安全之訓練課程，並有一系列之相關活動，以增進學生對核能發電的了解。

每年台電公司均配合全委會作業執行室於五月「中央防災週」至核電廠附近地區舉辦核子事故緊急應變計畫宣導溝通座談會。

(3) 有關訓練課程內容、期程、頻次等說明如本附錄C第6.1節〝緊急應變作業與支援人員訓練計畫〞。

(4) 上述(1)本公司承諾事項如審查問題編號P-C-001附表「龍門PSAR附錄C台電承諾事項一覽表」，#011)。

原能會審查(二)意見(ROCAEC Review Comments)：

完全沒有提到民眾宣導的規劃，並須要加進去。

台電澄清說明(Further Clarification)：

有關民眾宣導之規畫已於C.1.4.3.2(3)及安全分析報告審查意見編號P-C-001之再答覆中承諾說明。

編號(Track Number)： P-C-010

問題章節(PSAR Section)： 附錄C

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

報告中所用輻射單位宜全部改為新制SI單位。

問題答覆(Responses)：

同意遵照辦理。

原能會審查意見(ROCAEC Review Comments)：

請確實遵照辦理，因為某些新加的資料仍用舊單位。

台電澄清說明(Further Clarification)：

已遵照辦理，並已全部改寫如附錄C，所有輻射單位均改為新制SI單位。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： P-C-011

問題章節(PSAR Section)： 附錄C

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

報告中引用「準則」及其他資料時，未適當修正相關文字，致出現「本準則」、「詳如5.1.2.8要求」等字眼；C-93頁之近指中心通報作業亦不符規定。整體而言，本報告之編寫及台電公司自行審查之作業似嫌草率。

問題答覆(Responses)：

將切實檢討改進。

原能會審查意見(ROCAEC Review Comments)：

經查台電公司就P-C-014、P-C-018等問題所補充之資料中，仍有類似缺失，顯見台電公司並未如其答復辦理。台電公司應再詳細檢視PSAR附錄C(緊急計畫)所有內容，確實檢討改正。

台電澄清說明(Further Clarification)：

已詳細檢視PSAR附錄C所有內容並予確實改正。

編號(Track Number)： P-C-012

問題章節(PSAR Section)： 附錄C

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本附錄C應補充製作目錄，以便於查閱。

問題答覆(Responses)：

同意增列目錄，以方便查閱。

原能會審查意見(ROCAEC Review Comments)：

請台電公司於本附錄C版本修正確定後，於目錄之項次下加註各項內容之起始頁碼。

台電澄清說明(Further Clarification)：

已遵照辦理。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track　Number)：　　P-C-013

問題章節(PSAR Section)：　　附錄C　I.1　(新版附錄C　I.1)

初提日期(Question Date)：　　1997.12.15

問題內容(PSAR Question)：

本節應對電廠概況（如：電廠系統、容量、廠區位置、地理環境、廠區附近人口分布、道路狀況及氣象等）略加敘述。

問題答覆(Responses)：

所稱資料在本PSAR Chapter 1(Introduction and General Description of Plant)、Chapter 2(Site Characteristics)文中均有詳細說明，故建議無需在此處加以重覆敘述。

原能會審查意見(ROCAEC Review Comments)：

完全不同意台電公司的答復。附錄C(緊急計畫)為一獨立文件，未來演習或實際應變均將以其為依據，故應於此節摘述核電廠概況。此外，龍門計畫初期安全分析報告第一、二章係以英文書寫，恐不完全適用於緊急應變之參與人員。

台電澄清說明(Further Clarification)：

已遵照辦理，補充電廠概況說明如附錄C第1.1節〝電廠概況〞。

編號(Track Number)： P-C-014

問題章節(PSAR Section)： 附錄C I.2 (新版附錄C I.2)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節所述法規依據不完整。

問題答覆(Responses)：

同意於附錄C I.2適當之處補充增列完整之法規依據說明。

原能會審查意見(ROCAEC Review Comments)：

暫無進一步意見，請台電公司提出修正後之附錄C完整報告，以便整體審查其妥適性。

台電澄清說明(Further Clarification)：

已於附錄C第1.2節〝法規依據〞，補充增列完整之法規依據說明。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： P-C-015

問題章節(PSAR Section)： 附錄 C I.3 (新版附錄 C I.3)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節多為原則性說明，宜補充計畫架構摘要，並針對責任區分、組織分工等摘要說明。

問題答覆(Responses)：

有關計畫架構摘要、責任區分、組織分工等，在附錄C Ⅲ.1中均有相關之摘要說明，故建議無需在I .3中重複敘述。

原能會審查意見(ROCAEC Review Comments)：

本節名為「緊急計畫基本架構」卻無任何架構性說明，有名實不符之缺失；且若於第一節即闡述整體架構，當可使未來之使用者能迅速掌握整體大要，故請補充加強之。

台電澄清說明(Further Clarification)：

已遵照辦理，補充計畫架構摘要，並針對責任區分、組織分工等摘要說明如附錄C第1.3節〝緊急計畫摘要〞。

編號(Track Number)： P-C-016

問題章節(PSAR Section)： 附錄C II. 1 (新版附錄C 2.2)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節有關緊急計畫區的訂定方法不正確。

問題答覆(Responses)：

同意遵照修訂。

原能會審查意見(ROCAEC Review Comments)：

本節有關緊急計畫區的訂定方法不正確。

台電澄清說明(Further Clarification)：

已遵照辦理，補充核四廠緊急計畫區訂定方法說明如附錄C第2.2節〝緊急計畫區之說明〞。

編號(Track Number)： P-C-017

問題章節(PSAR Section)： 附錄C II. 1 (新版附錄C 2.2)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

有關緊急計畫區暫定為五公里之理由，請再加以詳細說明。

問題答覆(Responses)：

同意遵照修訂(說明同編號P-C-016之答覆)。

原能會審查意見(ROCAEC Review Comments)：

本節補充資料只述及核一、二、三廠之作法，請以此為基礎說明核四廠之作法。

台電澄清說明(Further Clarification)：

已遵照修訂，如附錄C第2.2節〝緊急計畫區之說明〞。

原能會審查(二)意見(ROCAEC Review Comments)：

C.2.2 完全沒有提到放射污染管制區的概念，必定會引起質疑，請加入說明。

台電澄清說明(Further Clarification)：

依據我國原子能法規及全國核子事故緊急應變計畫，並無〝放射污染管制區〞設置之相關規定。目前核能一、二、三廠亦未設置〝放射污染管制區〞。

編號(Track Number)： P-C-018

問題章節(PSAR Section)： 附錄C III. 1 (新版附錄C 3.1)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

各重要應變單位多未指明負責人及其在原單位之職稱。

問題答覆(Responses)：

擬於附錄C第3.1節〝廠內外緊急應變組織之編組與權責分工〞內容中增列緊執會部分各工作組之負責人及其所在原單位職稱，與廠內各緊急工作隊編組職務與原隸屬單位職稱對照表。

原能會審查意見(ROCAEC Review Comments)：

暫無進一步意見，請台電公司提出修正後之附錄C完整報告，以便整體審查其妥適性。

台電澄清說明(Further Clarification)：

已於附錄C第3.1節〝緊急應變組織之編組與權責分工〞內容中補充緊執會部分各工作組之負責人及其所在原單位職稱，與廠內各緊急工作隊編組職務與原隸屬單位職稱對照表。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： P-C-019

問題章節(PSAR Section)： 附錄C III. 1 (新版附錄C 3.1)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

廠內各緊急工作隊、組、中心負責人及其代理人、編組成員與人數等，應予明確規劃、說明。

問題答覆(Responses)：

1. 依目前工程進度，由於時程尚早，核四電廠組織與員額尚未規劃與確定，故對於廠內各緊急工作隊組、中心負責人及代理人、編組成員與人數等，目前尚無法明確規劃。
2. 本公司承諾將來在FSAR提出時，將就上述事項予以明確說明。

原能會審查意見(ROCAEC Review Comments)：

各緊急控制場所必須容納相關作業人員及設備，其編組人數如未能掌握，則如何規劃設計場所之大小？請台電公司依據核能四廠運轉需求及現有核能電廠運作狀況，保守規劃各緊急工作隊組應有編組人數，據以評估所需場所之大小。

台電澄清說明(Further Clarification)：

已遵照補充廠內各緊急工作隊、組、中心負責人及其代理人、編組成員相關說明如本附錄C第3.1節〝緊急應變組織之編組與權責分工〞。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： P-C-020

問題章節(PSAR Section)： 附錄C III. 1 (新版附錄C 3.1)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

表1-2中未說明緊急民眾資訊中心之任務；執行PASS取樣之工作隊組亦應予確定。

問題答覆(Responses)：

1. 緊急民眾資訊中心之任務在表20-1 (問題編號P-C-048)中已有明確說明。
2. 執行PASS取樣之工作隊組在表1-3 (問題編號P-C-018)中亦有明確說明。

原能會審查意見(ROCAEC Review Comments)：

表1-2中列有廠內各緊急組織之主要任務說明，獨漏緊急民眾資訊中心部分，請再予補正。另核能四廠PASS取樣分析作業究由緊急再入隊或緊急輻射偵測隊執行，亦請予規劃確定，而非檢附「準則」而已。

台電澄清說明(Further Clarification)：

已遵照增列緊急民眾資訊中心之任務說明，並補充執行PASS取樣之工作隊之編組任務說明如附錄C第3.1節〝緊急應變組織之編組與權責分工〞及表C.3.1-3。

編號(Track Number)： P-C-021

問題章節(PSAR Section)： 附錄C III. 2 (新版附錄C 5.1)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

未說明通知流程（包括：負責通知之單位與人員、通知之時機與內容等資料）。

問題答覆(Responses)：

本章節僅就緊急通知時機與作業流程作原則性之扼要說明，至於負責通知之單位人員、與內容等，屬於作業細節部分，本公司建議置於將來之核能四廠緊急計畫相關作業程序書中似較妥。

原能會審查意見(ROCAEC Review Comments)：

請說明未來核四廠緊急計畫相關作業程序書有那些，內容摘要為何，何時可定出。

台電澄清說明(Further Clarification)：

(1) 未來核四廠緊急計畫相關作業程序書及內容規定等如本附錄表C.1.4-1〝核能四廠緊急計畫章節之規定〞，表C.1.4-2〝緊執會緊急計畫實施程序內容之規定〞，及表C.1.4-3〝核能四廠緊急計畫實施程序內容之規定〞。

(2) 將於核四廠第一部機組初始燃料裝填前訂妥，並承諾如審查問題編號P-C-001附表「龍門PSAR附錄C台電承諾事項一覽表」，#002)。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： P-C-022

問題章節(PSAR Section)： 附錄C III. 2 (新版附錄C 5.1)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

第二類緊急事故時，廠內緊急組織之動員時機與方式（局部或全體動員）、局部動員之工作隊組等應予明確訂定，表2-1並應對應修正。

問題答覆(Responses)：

1. 表2-1係說明廠內各緊急作業中心在各種事故類別與情況下之動員時機，至於廠內各緊急工作隊組係分屬於各不同作業中心，其動員作業細節建議不在PSAR內詳述，本公司認為此部分置於核能四廠緊急計畫相關作業程序書中似較妥。
2. 擬另增列TSC、OSC、HPC、EPIC在第二類事故時之動員時機明確說明。

原能會審查意見(ROCAEC Review Comments)：

依Ⅲ.2節文字內容，第二類緊急事故時廠內緊急組織局部動員，與表2-1所述並不一致。因此，第二類緊急事故時廠內緊急組織之動員時機與方式(局部或全部)、局部動員之工作隊組等仍請釐清說明，並據以修正相關內容。

台電澄清說明(Further Clarification)：

已遵照於附錄CⅢ.2中明確訂定廠內緊急組織之動員時機與方式，即當發生第二類緊急事故時，所有廠內緊急組織應立刻全部動員。

編號(Track Number)： P-C-023

問題章節(PSAR Section)： 附錄C III. 3 (新版附錄C 2.1)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節請針對核四廠特性，補充說明各類事故之判定準則，如就核燃料、圍阻體等範圍，依儀表指示、系統狀況等訂定相關判定準則。

問題答覆(Responses)：

同意遵照辦理。

原能會審查意見(ROCAEC Review Comments)：

請針對核能四廠的狀況提出說明，不是將一些舊有的資料抄入。

台電澄清說明(Further Clarification)：

已遵照辦理，補充資料如本附錄C第2.1節〝緊急事故分類〞。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： P-C-024

問題章節(PSAR Section)： 附錄C III. 4 (新版附錄C 3.2)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

- 與醫院之協定中，附件一注重輻傷中心之建立，對緊急應變措施較無描述，附件二則只針對輻傷病患訂定急救方式，對一般傷患則付之闕如。
- 應補充提供與榮總簽訂之有效合約影本。

問題答覆(Responses)：

同意補充：

(1) 台灣電力公司/榮民總醫院核能電廠輻射傷害防治第二期計劃特約醫院委託合約。

(2) 台灣電力公司/國防醫學院(輻射傷害防治中心)籌建計劃特約醫院委託合約(含附約)

(3) 緊急醫療救護法(行政院衛生署84.8頒佈)

原能會審查意見(ROCAEC Review Comments)：

請台電公司就委託合約或法規所載事項之相關配合作業(如:負責通報連繫之人員、方式、作業要求等)，略作說明。

台電澄清說明(Further Clarification)：

已遵照補充相關之台電配合作業事項說明，如附錄C第3.2節〝廠外緊急應變組統之支援協定〞。

原能會審查(二)意見(ROCAEC Review Comments)：

C.3.2文中“係按廠界輻射程度....民眾防護措施”並不正確，請再重新寫過。

台電澄清說明(Further Clarification)：

已遵照辦理，修正如C.3.2。

編號(Track Number)： P-C-025

問題章節(PSAR Section)： 附錄C III. 4 (新版附錄C 3.2)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

與醫院的協定只適用於台北榮總，對位居貢寮之核四廠恐嫌太遠，宜仿核三廠在廠區附近找一適當之醫療單位，與其訂定初步急救協定。

問題答覆(Responses)：

發生核子事故時，針對一般傷患之緊急醫療救護：

(一)依八十四年八月行政院衛生署頒佈之「緊急醫療救護法」之規定，中央衛生主管機關(行政院衛生署)應會同中央消防主管機關(內政部)劃定緊急醫療救護區域，訂定緊急醫療救護實施計畫。省(市)、縣(市)政府應依此實施計畫辦理緊急醫療救護業務。因此當發生核子事故時，可動員依上述所建立之緊急醫療救護體系進行一般傷患之緊急醫療救護。

(二)核四廠所在地附近鄉鎮之醫療單位：

(雙溪、貢寮)

區分	醫院名稱	地址	連絡電話	備註
責任醫院	貢寮群體醫療中心	貢寮鄉朝陽街70巷8號	02-24941020	
協辦醫院	雙溪群體醫療執行中心	雙溪鄉新基南街19號	02-24931210	

(三)擬補充相關具體說明於本附錄C Ⅲ.4與Ⅲ.10內容中。

原能會審查意見(ROCAEC Review Comments)：

請台電公司再行評估是否應與核四廠廠區附近醫院簽訂初步急救協定或如何協助該等醫院提升輻傷初步急救能力，以建立完整有效之輻傷救治體系。

台電澄清說明(Further Clarification)：

核四廠區附近醫療單位對核四廠提供之緊急醫療支援說明如附錄C第5.13節〝廠外緊急醫療照護與輻傷救護措施〞，其內容涵蓋核四廠所在地附近鄉鎮之醫療單位(貢寮鄉：貢寮群體醫療中心、雙溪鄉：雙溪群體醫療執行中心)。

編號(Track Number)： P-C-026

問題章節(PSAR Section)： 附錄C III. 4 (新版附錄C 3.2)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節尚缺與軍方部隊之相關協定。

問題答覆(Responses)：

同意並補列於本節。

原能會審查意見(ROCAEC Review Comments)：

台電公司檢附之資料為核安二號演習資料，請針對核四廠提出與軍方部隊之相關協定。

台電澄清說明(Further Clarification)：

依據行政院頒佈之「核子事故緊急應變計畫」，全委會委員即包括國防部首長，且全委會下設三個作業中心一近指、救災、支援中心，其中支援中心由國軍部隊組成。按規定全委會中代表國防部委員之權責為負責國軍支援事項之命令下達與督導。當核四廠發生緊急事故，且廠界最大個人全身劑量率達0.2mSv/hr (20mr/hr)時，支援中心受命成立，並立即採取包括執行疏散區域內之交通管制及警戒、民眾疏運、輻射除污等救援行動。

編號(Track Number)： P-C-027

問題章節(PSAR Section)： 附錄C III. 5 (新版附錄C 5.3)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節尚缺兩種以上通訊方式之說明、各單位負責連絡人員及定期通訊測試計畫等。

問題答覆(Responses)：

1. 本公司認為核四PSAR中的緊急計畫章節部分，不同於緊急計畫作業程序書描述作業細節，故僅就原則性的要求(或承諾)加以敘述或說明即可，至於詳細之實施程序將在核四緊急計畫程序書中另行加以規範。
2. 擬建議如各單位負責連絡人員及定期通訊測試計畫等，於核四FSAR階段提出(編列於核四緊急計畫程序書中)。

原能會審查意見(ROCAEC Review Comments)：

即使目前沒有，也應該做承諾或者陳述作法，如：未來核四廠緊急計畫相關作業程序書有那些，內容摘要為何，何時可定出等。

台電澄清說明(Further Clarification)：

(1) 已遵照初審意見補充兩種以上通訊方式、各單位負責連絡人員及定期通訊測試計畫等說明如本附錄C第5.3節〝事故期間各項資訊傳遞〞。

(2) 未來核四廠緊急計畫相關作業程序書及內容規定等如本附錄C第1.4節〝核能四廠緊急計畫作業程序書摘要〞。將於核四廠第一部機組初始燃料裝填前訂妥，並承諾如審查問題編號P-C-001附表「龍門PSAR附錄C台電承諾事項一覽表」，#002)。

編號(Track Number)： P-C-028

問題章節(PSAR Section)： 附錄C III. 6 (新版附錄C 5.8)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

電廠須針對事故中緊急救援行動，如：傷患急救、設備搶修、事故評估及人員除污等依相關法規擬訂救援人員之緊急輻射曝露計畫，並指定或授權適當主管人員於事故中負責該等救援行動之核准與監督工作。

問題答覆(Responses)：

1. 本公司認為核四PSAR(緊急計畫)，不同於緊急計畫作業程序書，僅就原則性的要求(或承諾)加以敘述或說明即可，至於實施細節則將在核四緊急計畫程序書中另行規範。
2. 事故中緊急救援行動之污染管制行動程序、方法與平時相同，但平時污染管制僅實施於廠內，當意外事故發生時，廠區污染管制著重於人員進出管制，非必要之人員不得進入現場。
3. 其他要求事項，本公司承諾將在FSAR中予以明確規範。

原能會審查意見(ROCAEC Review Comments)：

不同意台電公司的答復。本題原意為要求擬訂緊急應變時救援人員最大容許劑量及其核准與監督體制等，請台電公司再作修正說明。

台電澄清說明(Further Clarification)：

已遵照增加針對事故中緊急救援行動，依相關法規擬訂救援人員之緊急輻射曝露計畫與指定適當主管人員負責該等救援行動之核准與監督工作等說明如本附錄C第5. 8節〝廠內輻射防護措施〞。

原能會審查(二)意見(ROCAEC Review Comments)：

C.5.8.2.2 文中(2)及(3)中的核四廠廠界是否指廠外，如果不是，應說明廠外相關之規劃與承諾。

台電澄清說明(Further Clarification)：

是指廠外，為免誤解，該文已修正如C.5.8.2.2。

編號(Track Number)： P-C-029

問題章節(PSAR Section)： 附錄C III. 6 (新版附錄C 5.8)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節未說明疏散人員清點方式及交通工具、傷患緊急救護措施與設備等。

問題答覆(Responses)：

1. 疏散人員清點方式及交通工具，涉及電廠的硬體設施種類、數量而有不同，擬建議於核四FSAR階段，各項規劃作業達到成熟階時編列於核四緊急計畫中。
2. 傷患緊急救護措施與設備之說明編列在附錄C Ⅲ.9中。

原能會審查意見(ROCAEC Review Comments)：

暫無進一步意見，請台電公司提出修正後之附錄C完整報告，以便整體審查其妥適性。

台電澄清說明(Further Clarification)：

已遵照初審意見補充廠內疏散人員清點方式及交通工具，傷患救護措施與設備等說明如附錄C第5.8節〝廠內輻射防護措施〞。

編號(Track Number)： P-C-030

問題章節(PSAR Section)： 附錄C III. 7 (新版附錄C 5.9)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節太簡略，請參照「核能電廠初期安全分析報告緊急應變計畫審查導則」擬訂較具體之措施。

問題答覆(Responses)：

同意補充本節〝附錄C Ⅲ.7 廠外民眾防護措施〞較具體之措施說明。

原能會審查意見(ROCAEC Review Comments)：

暫無進一步意見，請台電公司提出修正後之附錄C完整報告，以便整體審查其妥適性。

台電澄清說明(Further Clarification)：

遵照辦理。

編號(Track Number)： P-C-031

問題章節(PSAR Section)： 附錄C III. 8 (新版附錄C 3.2)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節除應參照Ⅲ.4節之審查意見補充說明外，並應補充分析所需軍、警、消防、交通等單位支援事項。

問題答覆(Responses)：

1. 廠外之軍、警、消防、交通等單位受全委會之命執行對廠外地區之緊急應變支援作業。
2. 同意遵照補充相關資料。

原能會審查意見(ROCAEC Review Comments)：

本節應分析核四廠本身可能需要軍、警、消防、交通等單位之支援事項，而非說明全委會辦理事項。

台電澄清說明(Further Clarification)：

(1) 廠外之軍、警、消防、交通等單位受全委會之命執行其對廠外地區之緊急應變支援作業。

(2) 依照核能電廠緊急計畫之規劃，各核能電廠應有能力應付事故時廠內緊急應變處理，如因故臨時需廠外緊急組織支援時，則循地方之救災體系，如撥〝119〞(公用電話)，請求地方政府迅速提供消防、警、軍、交通等單位協助事項。

(3) 目前三個核電廠(核一、二、三)均未與廠外之軍、警、消防、交通等單位另外簽定支援協定。

編號(Track Number)： P-C-032

問題章節(PSAR Section)： 附錄C III. 9 (新版附錄C 5.12)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節請補充說明初步傷患急救與除污之設備與能力

問題答覆(Responses)：

同意遵照辦理。

原能會審查意見(ROCAEC Review Comments)：

暫無進一步意見，請台電公司提出修正後之附錄C完整報告，以便整體審查其妥適性。

台電澄清說明(Further Clarification)：

遵照辦理。

編號(Track Number)： P-C-033

問題章節(PSAR Section)： 附錄C III. 10 (新版附錄C 5.13)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節請補充說明初步傷患急救與除污之設備與能力、核四廠附近醫院之名稱，以及傷患送醫之交通工具。

問題答覆(Responses)：

同意遵照辦理。

原能會審查意見(ROCAEC Review Comments)：

請補充緊急傷患急救時可容納多少傷患，有多少病床及相關急救設備等具體數字。

台電澄清說明(Further Clarification)：

已遵照辦理，補充說明如本附錄C第5.13節〝廠外緊急醫療照護與輻傷救護措施〞。

編號(Track Number)： P-C-034

問題章節(PSAR Section)： 附錄 C III. 11 (新版附錄 C 5.14)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節太籠統，請具體說明設備、人力及能量等資料。

問題答覆(Responses)：

同意補充附錄C Ⅲ.11〝簽約醫院接受治療傷患能力評估〞之設備、人力及能量等具體資料。

原能會審查意見(ROCAEC Review Comments)：

請將補充資料消化整理後再納入，目前整個附錄C似僅將各種文件放在一起而已，而非一份完整報告。

台電澄清說明(Further Clarification)：

已遵照初審意見補充具體資料說明簽約醫院治療傷患設備、人力及能量等如本附錄C第5.14節〝簽約醫院接受治療傷患能力評估〞。

編號(Track Number)： P-C-035

問題章節(PSAR Section)： 附錄C III. 12 (新版附錄C 6.1)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節請補充訓練課程內容、時數、頻次與考核方式等。

問題答覆(Responses)：

擬同意並補充有關訓練課程內容、時數、頻次與考核方式等說明。

原能會審查意見(ROCAEC Review Comments)：

本附錄C之內容若涉及其他單位時，請台電公司主動循正式程序協調定案後，再行納入。台電公司檢附全委會各中心、室訓練課程內容、頻次等資料作為本問題答復內容之一部分，是否已依此原則辦理，並請澄清說明。

台電澄清說明(Further Clarification)：

已遵照補充有關訓練課程內容、時數、頻次與考核方式等說明如本附錄C第6.1節〝緊急應變作業與支援人員訓練計畫〞。

編號(Track Number)： P-C-036

問題章節(PSAR Section)： 附錄C III. 12 (新版附錄C 6.1)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節應配合Ⅲ.8節，訂定廠外支援人員之訓練計畫。

問題答覆(Responses)：

同意遵照辦理。

原能會審查意見(ROCAEC Review Comments)：

本節應針對Ⅲ.8節(PC-C-31)分析結果，擬訂軍、警、消防、交通等支援人員之訓練計畫、課程、頻次等具體資料。

台電澄清說明(Further Clarification)：

已遵照訂妥如本附錄C第6.1節〝緊急應變作業與支援人員訓練計畫〞。

編號(Track Number)： P-C-037

問題章節(PSAR Section)： 附錄C III. 13 (新版附錄C 5.2)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節請補充負責通知之人員、通知內容、時機及所需時間等。

問題答覆(Responses)：

1. 擬同意增列負責通知之人員，及所需時間。
2. 緊急通知時機在附錄C Ⅲ.2中已有說明，故在此不再重述。
3. 至於通知內容，視事故程度之演變而有所不同，擬建議於Ⅲ.2〝緊急通知/動員時機與作業流程〞中適當之處予以補充。

原能會審查意見(ROCAEC Review Comments)：

答復的資料中未見所增列的內容。

台電澄清說明(Further Clarification)：

(1) 已遵照初審意見增列負責通知之關鍵性人員及所需時間如附錄C第5.2節〝通知各緊急應變組織及民眾所用方法與時間〞。
(2) 複審意見有關負責通知人員、通知內容、時機之說明已補充如本附錄C第5.1節〝緊急通知動員時機與作業流程〞。

編號(Track　Number)：　P-C-038

問題章節(PSAR Section)：　附錄C　III. 13 (新版附錄C　5.2)

初提日期(Question Date)：　1997.12.15

問題內容(PSAR Question)：

C-69頁，請以更直接的方式說明警報的內容、形式。

問題答覆(Responses)：

同意遵照辦理。

原能會審查意見(ROCAEC Review Comments)：

暫無進一步意見，請台電公司提出修正後之附錄C完整報告，以便整體審查其妥適性。

台電澄清說明(Further Clarification)：

遵照辦理。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： P-C-039

問題章節(PSAR Section)： 附錄C III. 14 (新版附錄C 5.10)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節應實際就龍門電廠緊急計畫區內人口、訪客、道路、交通狀況、氣象等資料進行疏散分析。

問題答覆(Responses)：

1. 因緊急計畫區內人口、訪客、道路、交通狀況等資料，其隨時間之成長性殊難逆料，在目前之PSAR階段即做此項分析似無太大意義，以其時效性而言，於FSAR階段提出似較符實際。
2. 進行此項疏散分析之前，有關緊急計畫區內人口、訪客、道路、交通狀況、氣象等資料均須事先加以調查、蒐集、統計、勘查、評估並加以規劃後，再進行該項疏散分析作業。
3. 本公司承諾將經過協調全委會就實際龍門電廠緊急計畫區內人口、訪客、道路、交通狀況、氣象等資料進行疏散分析，並將在FSAR中予以明確說明。

原能會審查意見(ROCAEC Review Comments)：

本節在PSAR提出時，如未能及時完成緊急計畫區內民眾疏散分析，則應於PSAR中訂定具體作法與時程，並承諾於FSAR中提出完整分析報告；另台電公司協調外單位(如：全委會)辦理事宜，應有往來書面資料以為佐證。

台電澄清說明(Further Clarification)：

(1) 因緊急計畫區內人口、訪客、道路、交通狀況等資料，其隨時間之成長性殊難逆料。在進行此項疏散分析之前，有關緊急計畫區內人口、訪客、道路、交通狀況、氣象等資料均須事先加以調查、蒐集、

統計、勘查、評估並加以規劃後，再進行該項疏散分析作業。

(2) 本公司承諾將協調全委會，就實際龍門電廠緊急計畫區內人口、訪客、道路、交通狀況、氣象等資料進行疏散分析，並在FSAR中明確說明。

(3) 上述本公司承諾事項如審查問題編號P-C-001附表「龍門PSAR附錄C台電承諾事項一覽表」，#006)。

原能會審查(二)意見(ROCAEC Review Comments)：

台電公司就執行緊急計畫區內訪客及居民疏散分析所承諾之時程，是否來得及配合全委會緊急應變作業流程？

台電澄清說明(Further Clarification)：

有關涉及核四廠外緊急計畫事項部分，本公司已於87年9月7日與全委會作業執行室經正式開會協調，將由全委會作業執行室主辦，台電公司協辦，逐步完成各主要相關廠外緊急計畫事項。有關本疏散分析之執行，將由台電公司於核四核燃料裝填前二年完成第一階段，包括核四廠緊急計畫區內人口分佈、遊樂設施與遊客動態等資料之調查與統計分析，提供作業執行室進行第二階段，包括殘障居民、交通流量、與道路狀況等評估分析(估計約需時1年)，以建立包括遊客在內之有效路網分析模式，全案將在核四廠核燃料裝填前半年完成。

編號(Track Number)： P-C-040

問題章節(PSAR Section)： 附錄C III. 15 (新版附錄C 5.11)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節請就疏散道路容量、緊急醫療能量等考量障礙內容、解決方式及期程等。

問題答覆(Responses)：

1. 疏散道路容量與緊急計畫區內人口、訪客、道路、交通狀況等資料有關，故本公司處理情形擬同前(P-C-039)。
2. 本公司承諾將經過協調全委會後就實際疏散道路容量、緊急醫療能量等資料考量障礙內容、解決方式及期程等，並將在FSAR中予以明確說明。

原能會審查意見(ROCAEC Review Comments)：

本節應依Ⅲ.14節(P-C-039)所訂時程，配合提出有關疏散道路容量、緊急醫療能量、收容站設置等相關障礙內容、解決方式及期程等。

台電澄清說明(Further Clarification)：

(1) 疏散道路容量與緊急計畫區內人口、訪客、道路、交通狀況等資料有關，故本公司處理情形同前(請參閱問題編號P-C-039之答覆說明)。

(2) 本公司承諾將協調全委會，就實際疏散道路容量、緊急醫療能量等資料考量障礙內容、解決方式及期程等，於FSAR中明確說明。

(3) 上述本公司承諾事項如審查問題編號P-C-001附表「龍門PSAR附錄C台電承諾事項一覽表」，#006)。

原能會審查(二)意見(ROCAEC Review Comments)：

本節應依C.5.10節(問題P-C-039)實際分析結果所發現之各項障礙內容分別提出解決方式及期程等。台電公司所承諾之時程，請再檢討。

台電澄清說明(Further Clarification)：

1. 有關核四廠外疏散道路之規劃，本公司已於87年9月7日與全委會作業執行室正式開會協調獲得共識，由全委會作業執行室主辦，台電公司協辦，委託學術機構按現有緊急計畫區內人口、訪客、道路、交通狀況等資料，就現有道路規劃出最佳路網疏散分析模式，估計完成疏散所需時間。
2. 有關緊急醫療能量，係由政府權責相關部門一行政院衛生署，在其全國緊急醫療網之架構體系下予以整體考慮。

編號(Track Number)： P-C-041

問題章節(PSAR Section)： 附錄C III. 15 (新版附錄C 5.11)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

- 無圖15-1。
- 應承諾將在FSAR提出時，明確標出可能使用的集結點。

問題答覆(Responses)：

1. 漏圖15-1，已予以補充及提供相關說明。
2. 本公司承諾將在FSAR提出時，經過協調全委會後明確標出可能使用的集結點。

原能會審查意見(ROCAEC Review Comments)：

1. 請在PSAR相關章節中做出具體承諾。
2. 看不出圖15-1的意義。請說明圖15-1與此章節中的描述有何關係。
3. 台電公司協調外單位(如：全委會)辦理事宜，應有往來書面資料以為佐證。

台電澄清說明(Further Clarification)：

(1) 已補充圖C.5.11-1，並增加相關說明。

(2) 本公司承諾將協調全委會後，在FSAR中明確標出可能使用的集結點。

(3) 上述(2)之本公司承諾事項如審查問題編號P-C-001附表「龍門PSAR附錄C台電承諾事項一覽表」，#012)。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track　Number)：　P-C-042

問題章節(PSAR Section)：　附錄C　III. 16 (新版附錄C　5.4)

初提日期(Question Date)：　1997.12.15

問題內容(PSAR Question)：

本節請大略說明構想中的評估設備、系統與方法，包括那些能力及必須滿足那些功能等。（參考PSAR附錄A）

問題答覆(Responses)：

同意遵照辦理。

原能會審查意見(ROCAEC Review Comments)：

本節答復資料為一般性說明，請針對其如何應用於核四廠補充說明之。

台電澄清說明(Further Clarification)：

已遵照補充說明如本附錄第5.4節〝廠內外輻射影響程度評估設備、系統與方法〞。

編號(Track Number)： P-C-043

問題章節(PSAR Section)： 附錄C III. 17 (新版附錄C 5.5)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節請大略說明構想中的評估設備、系統與方法，包括那些能力及必須滿足那些功能等。（參考PSAR附錄A）

問題答覆(Responses)：

同意遵照辦理。

原能會審查意見(ROCAEC Review Comments)：

暫無進一步意見，請台電公司提出修正後之附錄C完整報告，以便整體審查其妥適性。

台電澄清說明(Further Clarification)：

遵照辦理。

編號(Track Number)： P-C-044

問題章節(PSAR Section)： 附錄C III. 18 (新版附錄C 5.6)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節應具體承諾在FSAR中明確劃定輻射偵測路線及偵測點。

問題答覆(Responses)：

1. 附錄 C Ⅲ.18〝放射污染管制區內輻射偵測隊派遣說明〞據悉目前正由全委會委託學術機構研究規劃中。
2. 本公司承諾在FSAR提出時，將經協調全委會後明確劃定輻射偵測路線及偵測點。

原能會審查意見(ROCAEC Review Comments)：

請在PSAR相關章節中做出具體承諾。

台電澄清說明(Further Clarification)：

(1) 本公司放射試驗室已規劃於民國88年7月至90年6月間執行包括四廠在內之核一、二、三、四廠環境輻射監測計畫研發專案，將會針對核四廠環境輻射偵測路線及偵測點作規劃。

(2) 本公司將協調全委會後，在FSAR中明確劃定輻射偵測路線及偵測點並承諾如本附錄C第5.6節〝緊急計畫區內輻射偵測派遣說明〞。

(3) 以上本公司承諾事項如審查問題編號P-C-001附表「龍門PSAR附錄C台電承諾事項一覽表」，#013)。

編號(Track Number)： P-C-045

問題章節(PSAR Section)： 附錄C III. 18 (新版附錄C 5.6)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節雖提及空中偵測但並無具體作業程序。

問題答覆(Responses)：

已於本章節中適當位置補充空中偵測之相關具體作業程序。

原能會審查意見(ROCAEC Review Comments)：

暫無進一步意見，請台電公司提出修正後之附錄C完整報告，以便整體審查其妥適性。

台電澄清說明(Further Clarification)：

遵照辦理。

編號(Track Number)： P-C-046

問題章節(PSAR Section)： 附錄C III. 19 (新版附錄C 5.7)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節缺民眾訊息傳遞之說明。

問題答覆(Responses)：

同意補充民眾訊息傳遞之相關說明。

原能會審查意見(ROCAEC Review Comments)：

暫無進一步意見，請台電公司提出修正後之附錄C完整報告，以便整體審查其妥適性。

台電澄清說明(Further Clarification)：

遵照辦理。

編號(Track Number)： P-C-047

問題章節(PSAR Section)： 附錄C III. 19 (新版附錄C 5.7)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

請描述PADES-1的基本功能。

問題答覆(Responses)：

同意補充PADES-1基本功能之敘述(請參見問題編號P-C-046之答覆說明)。

原能會審查意見(ROCAEC Review Comments)：

暫無進一步意見，請台電公司提出修正後之附錄C完整報告，以便整體審查其妥適性。

台電澄清說明(Further Clarification)：

遵照辦理。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： P-C-048

問題章節(PSAR Section)： 附錄C III. 20 (新版附錄C 4.1)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節應依廠內各緊急控制場所之人員、設備、輻防等需求，明確劃定各場所設置地點及場所大小。

問題答覆(Responses)：

因核能四廠組織、員額尚未確定，似無法對各緊急控制場所之人員、場所大小予以明確規定，不過，在本節內容中對於各緊急控制場所應具有之設備與應有之功能、輻防等需求及各場所設置地點、場所大小容量等，均依相關法規予以原則性適切規範。

原能會審查意見(ROCAEC Review Comments)：

廠房等各必要場所應於建廠前即已設計、規劃妥當，否則如何建廠？本節請依廠內緊急控制場所之人員、設備、輻防等需求，明確規劃各場所設置地點與大小，並應保守規劃，以預留擴充餘裕。

台電澄清說明(Further Clarification)：

已遵照規劃各場所設置地點及場所大小如附錄C第4.1節〝廠內各緊急應變組織作業場所設置說明〞。

原能會審查(二)意見(ROCAEC Review Comments)：

1. 有關緊急應變場所的設置(如OSC、TSC等)應在C.P以前即規劃完成，不應在FSAR提出後才執行。
2. 台電公司規劃將緊急民眾資訊中心設於近指中心同一建物內， 地點似有不當，且近指中心目前亦未確定。

台電澄清說明(Further Clarification)：

1. 核四廠各緊急應變場所目前初步的規劃如下：

   TSC： 位於Switchgear Building 的地下一樓，面積約600平方米(181坪)。

   OSC： 位於冷修配廠房的3樓，面積約420平方米(127坪)。

   HPC： 位於冷修配廠房的3樓，面積約100平方米(30坪)。

   近指中心、緊急民眾資訊中心(EPIC)位於放射試驗室核四工作隊旁，係一單獨的建築物，二者佔地共約940平方米(284坪)。

2. 近指中心和EPIC雖座落在同一棟建築物內，但其功能和空間係完全分開，互不影響。

原能會審查(三)意見(ROCAEC Review Comments)：

C.1.4.2 TSC設置於開關場建築物之地下室是否合適，會不會對工作的執行有所影響，美國相關法規中是否有明確的要求，請說明。

台電澄清說明(Further Clarification)：

TSC位於開關廠房(Switchgear Building)地下一樓，緊臨汽機廠房靠近控制廠房符合NUREG-0696〝TSC的位置應距離Main Control Room(MCR)步行2分鐘〞之規定。TSC位於地下一樓，不影響工作之執行。TSC之位置關係圖如附。

原能會審查(四)意見(ROCAEC Review Comments)：

C.1.4.2 (3)中提到TSC之防震能力，必須符合我國一般建築防震要求，似乎不是很合理，國外的要求如何？核一、二、三廠目前的狀況為何，又一般建築的防震要求與核四OBE及SSE的差異為何，請說明。

台電澄清說明(Further Clarification)：

1. 各國對核能電廠之地震設計，均依其安全功能區分為不同之等級。
2. 依據美國NUREG-0696 §2.5技術支援中心(TSC)，結構要求如次 "The TSC complex must be able to withstand the most adverse conditions reasonably expected during the design life of the plant including adequate

capabilities for (1) earthquakes, (2)high winds (other than tornadoes), and (3) floods. The TSC need not meet seismic Category I criteria or be qualified as an engineered safety feature (ESF). Normally, a well-engineered structure will provide an adequate capability to withstand earthquakes. Winds and floods with a 100-year-recurrence frequency are acceptable as a design basis. Existing buildings may be used to house the TSC complex if they satisfy the above minimum criteria."。

依據「台灣電力公司緊急計畫準則」§5.1.2.3技術支援中心建築結構強度要求為：「技術支援中心建築應能承受其電廠壽限內最大之事故狀況，如(1)地震(2)強風(3)洪水等，但得不必符合如ESF(Engineered Safety Feature)般之防震一級的要求(Seismic Category I)，防震能力以符合我國建築技術標準防震要求，強風及洪水則以百年最大災害為設計基礎。」

3. 核一、二、三、四廠技術支援中心的耐震設計均係依據上述美國及我國規定進行設計。核能四廠技術支援中心耐震設計耐震分類為IIC級，其耐震設計值約為0.18G核能四廠的SSE為0.4G，OBE為0.2G 。
4. 核能四廠技術支援中心(TSC)地震設計係可對付回歸期475年的地震，再加上考慮建築物重要性後，又乘「用途係數」1.50而得出耐震設計值0.18G(按建築「用途係數」之範圍為1.0～1.5，核四廠技術支援中心已採用最保守之「用途係數」1.50以強化其地震設計。
5. 核能四廠萬一發生大於0.18G之地震且技術支援中心無法使用時，技術支援中心成員可轉移至控制廠房主控制室旁之參觀室繼續執行任務。

編號(Track Number)： P-C-049

問題章節(PSAR Section)： 附錄C III. 20 (新版附錄C 4.1)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節需要對緊急應變設施數據系統做更詳細的說明。

問題答覆(Responses)：

本節在緊急應變設施數據系統方面之說明(附錄C Ⅲ.20.B)，係參引NUREG-0696之主要規定：其中對於此系統之功能、資料內容、數據之儲存能力、可靠性與可用率、數據之核驗、... 等皆有作詳盡扼要之敘述。

原能會審查意見(ROCAEC Review Comments)：

暫無進一步意見，請台電公司提出修正後之附錄C完整報告，以便整體審查其妥適性。

台電澄清說明(Further Clarification)：

遵照辦理。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： P-C-050

問題章節(PSAR Section)： 附錄C III. 21 (新版附錄C 4.2)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節應依近指中心人員、設備及輻防需求，明確規劃設置地點及場所大小，並承諾會於FSAR明定近指中心地點。

問題答覆(Responses)：

1. 同意遵照補充相關說明。
2. 本公司承諾將在FSAR提出時，明定近指中心地點。

原能會審查意見(ROCAEC Review Comments)：

台電公司所檢附資料，將「近指中心如設於緊急計畫區內，應指定一後備場所」等文字刪除，與現有規定不符。本節仍請訂定時程，於建廠過程中依近指中心人員、設備及輻防需求，規劃核四廠近指中心設置地點及場所大小。近指中心如欲設於緊急計畫區內，並應設置後備場所。

台電澄清說明(Further Clarification)：

(1) 已遵照補充相關說明包括近指中心人員、設備及輻防要求，規劃近指中心場所大小如附錄C第4.2節〝近廠指揮協調中心設置說明〞。

(2) 本公司承諾將在近指中心地點規劃完成時，陳報原能會核備。

(3) 上述本公司承諾事項如審查問題編號P-C-001附表「龍門PSAR附錄C台電承諾事項一覽表」，#014)。

原能會審查(二)意見(ROCAEC Review Comments)：

- C.4-27頁中對於近指中心相關設置作了詳盡說明，但對於是否應設置後備近指中心仍未有明確承諾。

- C.4-27頁近指中心地點說明， 不符「核能電廠緊急計畫準則」近指中心應位於禁建區外，若設於緊急計畫區內，應指定一後備場所之規定；同準則中有關近指中心後備場所之設置要求亦未納入PSAR內容。
- FSAR提出時，近指中心是否設置完成，宜再澄清。

台電澄清說明(Further Clarification)：

由於核四近指中心按目前之初步規劃係設於廠界附近，緊鄰放射試驗室核四工作隊，故台電公司將在緊急計畫區外另設置一後備近指中心，並於核燃料裝填前完成。

其設施應符合準則要求如下：

1. 與原近指中心大小相當之作業空間。
2. 具備良好之對外通訊系統。
3. 具備電廠之重要技術資料。

原能會審查(三)意見(ROCAEC Review Comments)：

C.1.4.2 未提近指中心設置地點。

台電澄清說明(Further Clarification)：

已遵照辦理，修正如C.1.4.2。

編號(Track Number)： P-C-051

問題章節(PSAR Section)： 附錄C III. 22 (新版附錄C 6.2)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

在本節中，緊急應變計畫（編號24）部分，電力公司承諾將於核四廠第一部機組初始燃料裝填前，編妥緊急計畫實施程序與第一次全廠緊急計畫演習方案，並執行全廠演習。但所稱編妥全廠緊急計畫演習方案一語，仍嫌過於籠統，宜提出具體之廠內、外緊急計畫演習之程序書，以供於第一次緊急計畫演習前，了解核四廠緊急計畫之完備性，及其對已往歷次核能電廠演習上獲取之經驗是否已有回饋。

又，附件三上所提出之83年版「核子事故緊急應變計畫」，包括目錄一至九內之說明，僅為實施要點，並不足以取代核四廠緊急計畫演習作業程序書。

問題答覆(Responses)：

1. 本公司依據原子能法規第三十四條之規定，於核四廠申請核發核子反應器使用執照時，應於初次安放燃料前提報FSAR，其中包括核四緊急計畫。
2. 另本公司參考美國聯邦法規10CFR50.App.E. V之規定，至遲於核發核子反應器使用執照180天前，提出核四緊急計畫實施程序，使之足以說明本公司執行核四緊急計畫之能力。

原能會審查意見(ROCAEC Review Comments)：

請台電公司澄清說明〝核四廠緊急計畫實施程序〞是否包含本問題要求的廠內、廠外緊急計畫演習作業程序書。

台電澄清說明(Further Clarification)：

(1) 有關廠內緊急計畫演習程序書：

〝核四廠1424演練與演習程序〞內容包括各項廠內外演習時演習人員(全廠員工)所應採取之緊急應變措施。

緊執會現有之〝核能電廠緊急計畫演習評核/管制作業程序書(EP-CR-T、EP-ER-T)〞，係供廠內演習管制及評核人員遵行以進行演習評核與管制作業。

(2) 有關廠外緊急計畫演習程序書：

全委會另有程序書－演習評核與管制作業程序書可供廠外演習管制及評核人員遵循。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： P-C-052

問題章節(PSAR Section)： 附錄C III. 23 (新版附錄C 6.3)

初提日期(Question Date)： 1997.12.15

問題內容(PSAR Question)：

本節請補充歷次演習日期、地點、參與人數、經費及評核結果摘要等資料，且針對評核所發現之缺失亦請提出具體改善措施。

問題答覆(Responses)：

同意遵照辦理。

原能會審查意見(ROCAEC Review Comments)：

問題的答復為遵照辦理。但根據所附的內容來看，似未完全〝遵照辦理〞，本問題的重點為〝針對評核所發現之缺失亦提出具體改善措施〞，請對此再予答復。

台電澄清說明(Further Clarification)：

已遵照補充歷次演習結果與具體改善措施等相關資料如本附錄C第6.3節〝緊急計畫經驗回饋〞。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： F-D-001

問題章節(PSAR Section)： Appendix D

初提日期(Question Date)： 1997.11.10

問題內容(PSAR Question)：

1. 拆廠作業立即拆除英文為何是 DECON ？第一階段應是將電廠保持於安全貯存狀態，而不是安全停機狀態。
2. 大型商用反應器拆廠預估經費約為三億美元（1997 年幣值），須註明是否包含最終處置費。
3. 龍門計畫預定拆廠工作流程中燃料移出廠房之出處應該有所交待。
4. 後端營運費用基金收支保管及運用辦法中基金之動支範圍應包括後端營運工作之規劃研究、技術資訊評估分析、研究發展及溝通計畫，否則將來無法有效執行後端營運工作。

問題答覆(Responses)：

1. DECON 係 DECONtamination followed by dismantling 之縮寫； SAFESTOR 係 SAFE STORage 之縮寫； ENTOMB 係 ENTOMBment 之縮寫。
   同意將拆廠第一階段用詞由安全停機狀態改為安全貯存狀態。
   龍門計畫安全分析報告附錄D 第 D-13 頁倒數第十行，將修改為：
   "第一階段：移除用過核燃料及拆除電廠一般設施，將電廠保持於安全貯存狀態；"
2. 本附錄預估之大型商用反應器拆廠預估費用約為三億美元，係包括最終處置費用。
3. 根據目前台電公司用過核燃料最終處置計畫時程，處置場之運轉將可與核四廠拆廠時程銜接，故其拆廠時，移出廠房之用過核燃料，將可直接運送到最終處置場處置。
4. 貴會審查意見所建議納入後端基金之動支項目，台電公司皆有預算支應，故是否納入後端基金動支範圍，並不影響其執行成效。後端基金預定於 87 年 7 月 1 日起改制為經濟部所屬非營業基金，為因應此情勢，相關的後端基金保管及運用辦法正修訂中，台電所擬條文草案已將 貴會所建議項目納入後端基金之動支範圍。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： F-D-002

問題章節(PSAR Section)： Appendix D

初提日期(Question Date)： 1997.12.04

問題內容(PSAR Question)：

1. D-4 中指出龍門計畫之用過燃料池貯存容量，可容納未來運轉 40 年之所退出所有用過核燃料，但第 11 章 11-5-10 中之 radwaste building ventilation exhaust monitoring 中缺乏對 Kr-85 、 I-129 之放射性 range 之敘述，請說明或補充。
2. 在 D-4 中設計容量 20,000 桶低放射性廢料之貯存倉庫為 40 年貯存期，在這期間是否也要作低放射性廢料最終處置方案？
3. 40 年貯存期，應該對貯存設施、盛裝容器之塗料規格有所規定，以確保安全貯存。

問題答覆(Responses)：

1. The ranges of channel instrument for Kr-85 and I-129 are not specified in PSAR Section 11.5.2.2.7 since many vendors specify the sensitivity of their equipment in terms of Xe-131 for noble gas and iodine detection, respectively. Specific nuclides are not listed for some radiation monitors since the detection channel is typically looking for a large array of radioisotopes and not necessarily for a small specific set。
2. 廠內倉庫僅是供放射性廢料暫存，在廠內倉存期間，本公司仍將積極推動進行低放射性廢料境內與境外最終處置計畫。
3. 有關盛裝容器之規格，原能會相關法規已有明確規定，本公司將依原能會法規辦理。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track　Number)：　F-D-003

問題章節(PSAR Section)：　Appendix D

初提日期(Question Date)：　1997.12.09

問題內容(PSAR Question)：

1. 電廠拆除時各類廢料產量及處理情形，請參酌法規提出概要評估。
2. 電廠拆除時，用過核燃料管理規劃（假設最終處置場未能如期完時），請提出說明。

問題答覆(Responses)：

1. 核能電廠拆廠所產生廢料可分為中子活化廢料、污染性廢料及放射性廢料三類，其說明如下：
   A. 中子活廢料係指反應爐及其周圍生物屏蔽等於核能電廠運轉期間長期接受中子照射之機具設備，於拆廠時所產生廢料。
   B. 污染性廢料係指核能電廠運轉期間遭受放射性污染之機具設備，於拆廠時所產生廢料。
   C. 放射性廢料係指核能電廠拆廠所進行除污工作及廠內原有放射性廢液在排放前經過處理等作業所產生廢料。

   這些廢料將先經過減容及回收可再利用物料(例如金屬材料)等處理後，再將其包封後運送最終處置場處置。根據 NUREG/CR-0672 之拆廠廢料估計，核四廠每一部機組拆廠所產生上述三種拆廠廢料產量分別約為 296 立方公尺、22,380 立方公尺及 24,634 立方公尺，總計 47,310 立方公尺。
2. 若用過核燃料最終處置場無法如期在核四廠拆廠之前運轉，可比照核一、二廠模式，興建中期貯存設施貯存；或者屆時若再處理已屬可行，亦可運送到國外再處理，以爭取緩衝時間完成最終處置場。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： F-D-004

問題章節(PSAR Section)： Appendix D

初提日期(Question Date)： 1997.09.30

問題內容(PSAR Question)：

1. 依據1990年日本原子力學會誌(Vol, 32, No, 5) 所載「改良型沸水型原子爐(ABWR)之開發及實用化」一文之第Ⅱ章「ABWR之開發技術」第10節「廢棄物發生量低減技術」(p. 434)所述，固體廢棄物發生量每年僅為100桶以下。
2. 「參、核廢料處置計畫」之「一、(一)境內處置」提及世界上已完成低放射性廢料處置設施之國家，請增列日本青森縣六個所村處置設施。

問題答覆(Responses)：

1. 後端營運主要是負責放射性廢料產生後之下游處理與處置工作，本附錄所預估核四廠運轉每部機組每年產生250桶低放射性固化廢料，係引用龍門計畫採購規範之相關規定，作為規劃低放射性廢料最終處置場之設計容量依據。
2. 納入。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： F-D-005

問題章節(PSAR Section)： Appendix D

初提日期(Question Date)： 1997.11.11

問題內容(PSAR Question)：

低放射性廢料境內最終處置計畫之五個階段，請列出時程，並以甘氏圖表示其進度及關鍵途徑。

問題答覆(Responses)：

經濟部已於八十六年十二月廿二日以經(86)國營字第八六五四三七四號函，檢送「台電公司『低放射性廢料最終處置』案具體方案與時程之說明」，請 貴會酌處彙陳行政院。低放射性廢料最終處置計畫之時程說明及甘特圖如附。

低放射性廢料最終處置計畫時程

本計畫各階段所需作業時間與建議場址所具備之條件如位於本島或離島、處置方式採用淺地層方式（平坦地形）或隧道處置方式（合適之山坡地形）…等息息相關。基於目前場址評選累積之經驗，本島場址採用隧道處置方式之機會較高，而離島場址採用淺地層處置方式之機會較高；且均需興建專用運輸碼頭之機會較高；僅就上述假設條件預估未來時程如后(其他說明詳如次頁備註)：

階段名稱	離島場址（無碼頭）		本島場址（無碼頭）	
	作業時間	作業期間	作業時間	作業期間
第一階段：場址選擇及處置方式評估（含地質調查）	三十六個月	八七、六｜九〇、五	三十九個月	八七、六｜九〇、八
第二階段：環境影響評估（含環境調查）	三十六個月	八七、六｜九〇、五	三十九個月	八七、六｜九〇、八
政府主管機關審查「可行性研究報告」及「環境影響說明書	二十二個月	九〇、六｜九二、三	二十二個月	九〇、九｜九二、六
第三階段：工程設計分析及場址精查	三十五個月	九二、四｜九五、二	三十九個月	九二、七｜九五、九
政府主管機關審查「建造許可申請書」及其他相關文件	十三個月	九五、三｜九六、三	十三個月	九五、十｜九六、十
第四階段：施工	五十二個月	九六、四｜一〇〇、七	七 十個月	九六、十一｜一〇二、八
政府主管機關審查「運轉許可申請書」及其他相關文件	六個月	一〇〇、八｜一〇一、一	六個月	一〇二、九｜一〇三、二
第五階段：運轉、封閉與監管	一〇一、二開始運轉接收處置低放射性廢料		一〇三、三開始運轉接收處置低放射性廢料	

備註：一、在未進行建議場址之鑽探、地物、地化等相關場址調查前，上述預估時程具有較大之不確定性，於完成本計畫第一階段之「可行性研究報告」，可提供更準確之未來時程。

二、前表所示第一階段及第二階段可以同時（平行）作業，所需作業時間係以開始進行場址之地質及環境調查工作為起算基準。第二階段之「環境影響說明書」須併同第一階段之「可行性研究報告」陳報經濟部。

三、第二階段「環境影響說明書」經陳報政府機關審查後，若不需進入「環境影響評估法」所稱之「第二階段環境影響評估」時，所需作業時間可減少十五個月。

四、第三階段「工程設計分析及場址精查」及第四階段「施工」所需之作業時間包含「發包作業」所需時間（六個月），即招標文件準備及其內部審查核准作業(一個月)、公告期（二個月）及審決標作業（三個月）。

台電公司低放射性廢料最終處置計畫主時程(本島場址、無碼頭)

計畫時程	開始日期	完成日期	期間(月)	87	88	89	90	91	92	93	94	95	96	97	98	99	100	101	102	103
第一階段： 場址選擇/處置方式評選(含地質調查)	87/06	90/08	39																	
第二階段： 環境評估	87/06	90/08	39																	
政府有關機關審查 ・可行性研究報告 ・環境影響說明書	90/09	92/06	22																	
第三階： 段場址精查／工程設計	92/07	95/09	39																	
政府有關機關審查 .建造許可申請書 .其他相關文件	95/10	96/10	13																	
第四階段： 施工	96/11	102/08	70																	
政府有關機關審查 .運轉許可申請書 .其他相關文件	102/09	103/02	6																	
第五階段： 運轉	103/03																			

台電公司低放射性廢料最終處置計畫主時程(離島場址、無碼頭)

計畫時程	開始日期	完成日期	期間(月)	87	88	89	90	91	92	93	94	95	96	97	98	99	100	101	102	103
第一階段： 場址選擇/處置方式評選(含地質調查)	87/06	90/05	36																	
第二階段： 環境評估	87/06	90/05	36																	
政府有關機關審查 ・可行性研究報告 ・環境影響說明書	90/06	92/03	22																	
第三階： 段場址精查／工程設計	92/04	95/02	35																	
政府有關機關審查 .建造許可申請書 .其他相關文件	95/03	96/03	13																	
第四階段： 施工	96/04	100/07	52																	
政府有關機關審查 .運轉許可申請書 .其他相關文件	100/08	101/01	6																	
第五階段： 運轉	101/02																			

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： D-006

問題章節(PSAR Section)： Appendix D

初提日期(Question Date)： 1998.05.12

問題內容(PSAR Question)：

目前國外對於後端營運有採用可監測回收之貯存方式(Monitored Retrievable Storage)，在本章報告中是否列入考慮，抑或將來需要時再另行送審，請澄清。

問題答覆(Responses)：

國外所採 MRS 貯存方案，係因應廠內用過核燃料池儲存容量不足提供額外儲存容量之需。就龍門計畫而言，因廠內用過核燃料池之儲存容量可滿足電廠運轉四十年間之儲存需求，故不需考量增加如 MRS 之額外貯存方案。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： E-001

問題章節(PSAR Section)： Appendix E

初提日期(Question Date)： 1998.01.07

問題內容(PSAR Question)：

一、 經驗回饋項目編號P-12「廠房空間、管路配置設計應考慮維修與通行之便利性」是否不只限於管路配置，而對其他如閥類配置、電纜支架(Cable Tray)配置等之維護檢查、運轉巡視及人員通行便利性等亦作整體性考量。

二、 海水管路之配置是否針對腐蝕防護對策及維護檢查之便利性等作考量?

問題答覆(Responses)：

一、 依據核四廠核反應器暨附屬系統採購規範附錄A第6章第4.2節，有關廠房設計及佈置之規範，已涵蓋：廠房及設備配置、系統（管路、電纜托架等）支架配置、管路（含管閥、管路配件等）配置、電氣（含電纜托架、導線管等）等等之佈置要求，故經驗回饋項目編號P-12，不只限於管路配置，對其他如閥類配置、電纜托架配置等之維護檢查、運轉巡視及人員通行便利性等亦有整體性考量。

二、 根據PSAR3.7.3.12，所有地震分類1級之埋管均安裝於Tunnel或Trench內，故核四SSW已設計安裝於trench內；另依核四廠核反應器暨附屬系統採購規範附錄A第6章第 4.2.4.1節之要求，已考慮維護檢查之便利性。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： E-002

問題章節(PSAR Section)： Appendix E

初提日期(Question Date)： 1998.01.07

問題內容(PSAR Question)：

國外已有核能電廠拆廠經驗及相關研發建議，是否在核四設計上有參考回饋之處?

問題答覆(Responses)：

核四廠核反應器設計是採用美國奇異公司進步型沸水式核反應器(ABWR)為藍本，奇異公司是美國首座大型核能機組Shippingport拆廠之Decommissioning Operations Contractor，因此在設計ABWR時，即已充分考慮Shippingport之拆廠經驗，預留足夠之餘裕。附件即奇異公司提供之補充說明。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Attachment of ROCAEC Track No. E-002

Decommissioning Experience

GE, as the Decommissioning Operations Contractor, was responsible for decommissioning the Department of Energy (DOE) owned Shippingport Atomic Power Station. The Shippingport Atomic Power Station was the first large scale, commercial central station nuclear power plant constructed in the United States, and the first full size nuclear power plant to be decommissioned. It operated from 1955 to 1982, when it was shutdown for testing. After two years of testing, the fuel was removed, and the plant turned over for decommissioning in late 1984. The decommissioning was completed in January,1990, more than six months ahead of schedule. The work was performed safely and cost effectively. Techniques and processes employed by GE resulted in personnel radiation exposure accumulation at less than 20% of those estimated by DOE. The DOE returned the site to the Duquesne Power Co. for its unrestricted use. The decommissioning of Shippingport Atomic Power Station was done using standard tooling, equipment and demolition methods coupled with careful planning and innovative methods to minimize workers' exposure and meet budget requirements. The project was completed at $1 million below targeted cost. It is anticipated that additional nuclear plants will have been decommissioned during the forthcoming 65 years, and that significant advances, as yet undefined, will be made in the tooling and methods for decommissioning.

Enhancement of Accessibility

Various elements of the ABWR design enhance the accessibility of personnel, equipment (contaminated and non-contaminated), and machinery during plant operation and maintenance. These same features support the decommissioning activity. Special attention has been given in the ABWR design to provide access and equipment handling capabilities for the maintenance of the major components. Overhead monorails, or similar lifting devices, are installed for the removal of valves, pumps and heat exchangers. The design and location of these handling devices, installed to ease the removal of the equipment for maintenance, are closely coordinated with access hatches for equipment removal. Piping is laid out to avoid the need to remove piping for equipment removal. Studies have been made which show the building and equipment layout affords space to transport the removed equipment, out of the building if necessary, for maintenance. Electrical and control panels and cabinets are located so they can be readily removed. The ABWR is, in fact, designed to provide access to equipment for its inspection and maintenance without the need of special access equipment, such as temporary scaffolding or ladders.

Decommissioning is greatly simplified because of the ABWR's emphasis on ease of equipment removal and accessibility:

- Equipment removal can proceed without demolition of structures, or the need to provide special handling equipment.
- The existence of coordinated lifting/handling equipment, hatches and access routes during plant operation will make it possible to benefit, during decommissioning, from the experience and procedures used to remove the equipment during plant maintenance.

- Because equipment can be removed without removing piping, removal of piping and equipment can be decoupled from each other, to afford more decommissioning flexibility.

The ABWR is designed to meet the intent of Regulatory Guide 8.8 (ALARA). The same features that reduce personnel exposure during operation serve to reduce decommissioning worker exposure:

- Clean, uncontaminated areas are maximized. More locations and equipment will be able to be removed without expensive and time consuming radiological controls.
- Contaminated areas are better defined, contained, and limited. This restricts the amount of areas that will require radiological controls restrictions and personnel protective measures, thus enhancing productivity. It also minimizes the amount of contaminated waste generated.
- High dose rates associated with external recirculation systems are a major contributor to the exposure in current BWRs. This external system has been eliminated in the ABWR. Decommissioning workers will not be exposed to it.
- Flushing, draining, and decontamination provisions incorporated in the ABWR design will permit periodic reduction if necessary of piping dose rates during operations. These same systems can be used to reduce contact and general area dose rate levels in preparation for decommissioning.
- Materials used for components subjected to radiation have been selected to limit the radioactivity from neutron activated products. There will be less dose to decommissioning workers, and less Curie content to deal with at the time the plants are to be dismantled. It is possible to process some of these materials to reach levels of radioactivity that are below regulatory concern.
- Implementation of optimized water chemistry recommendations will minimize the inventory of corrosion products during plant operation. This reduced inventory will also benefit decommissioning activities.

These design elements, along with the material application practices discussed later, directly enhance the ability to decommission the ABWR.

Optimization of Material Selection

The material selection for the metallic components subject to radiation effect considered the reduction of radioactivity from neutron activated products. However, the characteristics of the material selected also meet other requirements in the Bid Specification and will not adversely reduce neutron economy nor cause any detrimental effects on reactor operation.

Material application for piping, tubing, vessel internal surfaces and other components that come in contact with radioactivity is discussed below. Low cobalt and nickel contents have been specified wherever possible. As noted above, these low levels minimize the inventory of activated corrosion products. This reduces the contact and general area dose associated activated corrosion film layer on these components. It also reduces the potential of this material collecting in a location and causing "hot spots" that contribute to dose during plant operation as well as decommissioning. Stainless steel is used in portions of the system such as the reactor internal components and heat exchanger tubes where high corrosion resistance is required. The nickel content of the stainless steels is controlled in accordance with applicable ASME material specifications and is typically in the range of 9 to 10.5%. The cobalt content is controlled to less than 0.05% in the XM-19 alloy used in the control rod drives.

Ni-Cr-Fe alloys such as Alloy 600 and Alloy X-750, which have high Nickel content must be used in some reactor vessel internal components, since no suitable alternative low Nickel material is available. An example of this need would be locations that require materials with special thermal expansion characteristics along with adequate corrosion resistance. Note that the cobalt content in the Alloy X-750 used in the fuel assemblies is limited to 0.01% in the active core. The high cobalt alloy, Stellite, is only used for hard facing of components which must be extremely wear resistant and for which no suitable alternative exists. An alternative material (colmonoy) has been used for some hardfacings in the core area.

Carbon steel is used in a large portion of the system piping and equipment outside of the nuclear steam supply system. This material is typically low in Nickel content and contains a very small amount of Cobalt activity.

Prevention of Concrete Contamination

The application of appropriate coatings on the concrete surfaces to reduce the complexity of the future decontamination and decommissioning is considered in the ABWR design. Epoxy-type wall and floor coverings have been selected which provide smooth surfaces to ease decontamination surfaces. Expanded metal-type floor gratings are minimized in favor of smooth surfaces in areas where radioactive spills could occur. These coatings have been demonstrated to be effective in currently operating plants. Equipment and floor drain sumps are stainless steel lined to reduce crud buildup and to provide surfaces easily decontaminated.

All the concrete surfaces that are exposed for long periods to potentially contaminated water are lined with stainless steel. The steel liners avoid concrete contamination, enhance the ability to maintain water clarity and provide easily decontaminated surfaces. The lined areas are: the Spent Fuel Storage pool, the Dryer/Separator pool, the New Fuel Vault and the Suppression Pool. In addition, the containment (wetwell and drywell) and the reactor pedestal and shield walls within the containment are lined with carbon steel. This prevents this material from being contaminated as a result of operational leakage. Additionally, this material does not contain reinforcement bars, which if present would become activated. The absence of the rebar enhances dismantling.

It should be noted that the Shippingport plant had large concrete spent fuel, reactor cavity and equipment storage pools that were largely unprotected throughout the 20+ years of plant operation. Even under these conditions, core sampling showed that contamination had penetrated only about 3.2 mm into the surface of the concrete. In those areas where cracks had formed, the activity migrated along the crack opening, but again, only penetrated about 3.2 mm into the adjacent surface. Standard concrete scabbing techniques, with appropriate engineering controls to contain the contaminated concrete dust permitted the bulk of this material to be released for unrestricted use. Therefore, should any unprotected concrete area become inadvertently contaminated, the effort to recover the area and the waste generated should be minimal.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： E-003

問題章節(PSAR Section)： Appendix E經驗回饋C-07

初提日期(Question Date)： 87年2月3日

問題內容(PSAR Question)：

編號C-07「廠區建築物防風設計之改良」，執行現況述及已在採購規範規定一級耐震廠房的設計風速為70 M/S，其他所有廠區內建築物的設計風速為54 M/S；此與PSAR 3.3節設計風速194.4 Km/h (54 M/S)，設計颱風風速252 Km/h (70 M/S)之說明有所差異，請澄清。

問題答覆(Responses)：

核四廠一級耐震廠房的設計已考慮本地強烈颱風的破壞力，設計風速符合採購規範的要求。說明如下:

在核四採購規範3.3.5.1.2節中規定，核四廠設計要考慮正常設計風速及極限設計風速，並規定正常設計風速為每秒54公尺，極限設計風速為每秒70公尺。

奇異公司在PSAR 3.3.1.1節中說明，設計風速為每秒54公尺，在3.3.2.1節中說明設計颱風風速為每秒70公尺，在3.8.4.3.1.1節; 3.8.4.3.1.2節及3.8.4.3.1.3節中也分別說明設計會考慮每秒70公尺的極限設計風速並說明其負載組合方式。