審查意見摘要表

（第十章 ～ 第十六章）

第十章　蒸汽與動力轉換系統

第十一章　放射性廢料管理

編號	內　　　容
R-11-01	PRM 系統應有專節做完整描述，所有 monitors 訊號每隔固定週期均傳送至控制室的 PCS 貯存起來。
R-11-02	有關 sampling 部分描述太簡略，除取樣目標外，取樣位置、方法、裝備等亦應交待。
R-11-03	請說明 tritium 的監測及其 release 量如何來評估。
R-11-04	核四廠之運轉乃多年以後之事，核四廠整體之輻防設計應留有足夠之 margins 以因應日後法規之修改。
R-11-05	我國游離輻射防護安全標準早已使用新單位，未來監測儀器全為新購，故不應只是舊儀器單位的轉換而已。
R-11-06	表 11.5-4 第四欄中所使用單位，應與我國游離輻射防護安全標準第四表的單位一致。
R-11-07	表 11.5-5 第三欄中的文字：Gross α&β Tritium，列在同一列是否錯誤？又是否確定為α？
R-11-08	表 11.5-6 第四欄中 Gross alpha 的值是否偏高？
R-11-09	表 11.5-7 第二、三、四欄中的文字：As above，語意不清，應修正。又第三欄中 Gross alpha 之後的符號，不知代表什麼？
R-11-10	請說明核四廠與以往之 BWR 電廠在 Source terms、gaseous 與 liquid effluents 有何差異，並說明其 design bases 之依據。
R-11-11	請說明何以表 11.5-2 不含 Kr-85，及 Kr-85 之偵測方法。
R-11-12	表 11.5-1 中，為何所使用的單位有些是 mSv/h，有些則是 cpm？
R-11-13	表 11.5-2 及 11.5-3 Expected Activity 的值是參考那些電廠經驗？日本柏崎電廠 ABWR 機組的經驗是否有考慮？
R-11-14	請說明為何有些系統只考慮劑量率 mSv/hr 而不考慮 I-131 及 particulates？請說明各系統所考量的依據？
R-11-15	請分別說明 high-high、high、downscale、inoperative trip 發生後所應採取的措施。
R-11-16	第 11.5.3.1 節述及要符合 10 CFR 20 Limits，類似問題應

編號	內　　容
	先考慮本國法令才是。
R-11-17	第 11.5.3 節提及 Liquid and Gaseous releases are monitored for gross gamma radioactivity，只要 monitor gross gamma 即可嗎？
R-11-18	第 11.5.5.4 節，請說明為何 Audits and verification during normal plant operation are out-of-scope for the Lungman NPS？
R-11-19	表 11.5-6 對於每三個月分析乙次之核種(如 Sr-89、Sr-90)是否會有活度甚高但廢水已排出廠外之可能？若有應如何改善？
R-11-20	有關「廠外輻射監測計畫」意見如下： (1)依本會「環境輻射偵測規範」，核設施在運轉前之調查須實施二年以上，俾使核設施於運轉後執行環測時可熟練運作，為此若於 FSAR 提出是否時程上將不相符。 (2)運轉前的背景調查在於取得有關之關鍵性核種、關鍵性曝露途徑及關鍵群體等資料，作為運轉後環測計畫的擬定及評估劑量的參考資料，而有關此類調查結果見諸於何章節，或其調查將於何時進行。 (3)為了解核設施運轉後之影響，更需有運轉前環境輻射及環境試樣的背景強度及特性。
F-11-01	1.澄清重要結構物耐震設計是否同時考慮水平及垂直加速度？
	2.澄清放射性氣體廢料統滯留管活性炭床之設計。
	3.澄清活性炭床上游是否設計有預濾器。
F-11-02	1.說明在起動運轉階段是否有機械真空泵之類似設計。
	2.反應爐冷凝水過濾除礦系統是否採中空纖維式過濾器。
F-11-03	1.澄清 I-131 之一致性 11.1Core release rate。
	2.每年每部機組 3700MBq 液體排放，40 年運轉放射性物質是否會累積在排放口附近。
	3.經由各個煙囪的排氣係以什麼設備處理並未交待。
	4.200 多噸 charcoal 多久須更換。

編號	內　　　容
	5.11.3.9.2 ventilation release 50C 是何意？
	6.澄清 25℃時 Kr 及 Xe 之吸附係數。
	7.11.3.2.1 中為何在 38℃。
	8.Table 11.3-3 中 Xe-131m 5.5E-01 MBq/s 與 Table 11.1-1 中數字不一致。
F-11-04	1.澄清濕性及乾性廢料固化前之核種分析及輻射劑量分析
	2.澄清廢料桶表面擦拭試驗、劑量率及核種濃度測量及貯存設施自動偵檢設備設計。
	3.固化劑貯存小於 25Gy/hr 之低劑量區澄清。
	4.Figure 11.4-1 secondary path 之說明及使用，何時使用亦未交待？
F-11-05	1.評估放射性氣體及液體稀釋狀態使用之電腦模式，在台灣使用之實用性與準確度驗證。
	2.廢料容器系統除 55 加侖桶外，是否規劃採用其他型式，請參酌法規提出補充說明。
	3.澄清廢樹脂之處理原則(減容或固化)。
	4.固化系統之及時 QA/QC 之控制能力。
F-11-06	1.說明圖 11.4.1 液體流程各液體流量成分，及後續處理？
	2.澄清 wet waste volume reduction、焚化、壓縮、RWCU filter sludge settling 產生廢料情形。
	3.澄清 LPW 之 hours to process Max.daily rate。
	4.圖 11.2-2 及 11.2-3 部分處理程序未定，請提出較詳細之處理流程設計、產物品質預估等資料。
F-11-07	1.請說明洗衣廢水處理後回收再使用的可行性。
	2.說明各廠房之 activity fraction。
F-11-08	固化系統應考量使用較高效率之固化減容方式。
11-29	澄清「流程與排放之輻射監測與取樣系統」是否考慮「事故後取樣系統」(PASS)。
11-30	補充在減廢方面採取的步驟(含用過樹脂是否焚化)。

第十二章　輻射防護

編號	內　　　　容
12-001	表 12.2-7 及相關表中的 class 分類代表什麼應說明之。
12-002	表 12.2-18a 之 Rotating-ball Spindle 列中，為何 Before Cleaning 的數值為零？又 Throttle Bushing 列中，為何 Before Cleaning 與 After Cleaning 的數值相同？
12-003	表 12.2-20 及相關表中的 Maximum Technical Specification 代表什麼，應說明之。
12-004	表 12.2-21 之計算所用的程式(計算 T body 及 Kidney)，並不符合 10 CFR Part 20 及 ICRP-30 的需求。
12-005	表 12.2-23 中計算的是 internal dose，為何用 annual dose(mSv/yr)，而非 committed dose(mSv)？
12-006	表 12.2-30 中右下角，Totals 的數值 3.8E+01 錯誤。
12-007	本章所用之劑量單位很亂(有用到 Sv， Gy， rem，及 R)，所有之單位請使用 SI 制。
12-008	表 12.5-2 中 Tritium Counting 的 Qty 為多少？
12-009	表 12.5-4 中 Pen Dosimeter 數量較多，Alarm Dosimeter 數量較少，似不符合當前的發展趨勢。
12-010	表 12.5-7 及 12.5-8 中，mr 應改用 mSv 表示？
12-011	第 12.3.1.3 節在 PSAR 管制地區自 0-1 mSv/h 共分 6 區，此與台電現行的做法，是否一致？
12-012	表 12.5-2 Laboratory Equipment MCA 偵檢頭之規格可能有錯誤，請再檢討。
12-013	表 12.2-8 至表 12.2-12 source term 之來源及 uncertainty 有多大？各核種 class 分類依據為何？表 12.2-18 如何得到，不準度如何，請說明。
12-014	第 12.2.1.2.7 節之 source data 未提供。
12-015	第 12.2-6 頁 Reactor Startup Source 是那種射源？
12-016	表 12.2-20 為一部機組或兩部機組之數值。max. tech. spec. 與 annual average 間有何關係？其中 Sr-89 此二值似有異常。
12-017	表 12.2-21 part A 及 part C 是否為體外劑量。

編號	內　　容
12-018	Reactor Pressure Vessel 及生物屏蔽 concrete 中是否應做特殊元素含量限制考慮？
12-019	第 12.3.13 節 Radiation Zoning 劑量率範圍與表 12.5-1 不一致，是否應有一致之劃分？
12-020	第 12.3.2 節 Shielding 中所用分析程式工具及核數據均為 30 年前老舊之技術產品，建議核四廠之 FSAR 應採用 update 之程式及數據。
12-021	第 12.3.4 節 Monitoring instrumentation radiation monitoring system 都應要求使用電腦連線系統。
12-022	1.第 12.3 節廠界體外劑量計算並未說明。 2.第 12.3 節大型廢料貯存設施造成界劑量並未估算考量。 3.第 12.4 節廠界 direct 及 skyshine radiation dose 影響並未評估說明。
12-023	第 12.5 節 表 12.5-2 至表 12.5-4 所提資料太老舊，應參考核能二廠目前所使用設備加以修正更新。
12-024	法規的引用應以我國法規為第一優先，且同時符合美國相關法規，然本章中並未引用到我國的法規。
12-025	台電現已有三座核能電廠，與運轉有關之輻射防護政策，及如何將運轉經驗回饋至核四之設計等承諾均應在 PSAR 中提出。
12-026	第 12.2.2 節 請問在那份文件可以看到工作人員對空浮放射性物質之曝露可符合 10CFR20 規定的直接評估資料。
12-027	在大於 1mSv/h 的區域會上鎖防止人員不當進入，請問在控制室或保物管制站是否會有訊號可看出該區域有人員不當進入？
12-028	表 12.4-1，(1)反應器廠房工作項目中漏掉第 12.4.1 節中所述及之工作；(2)在 Turbine building 中 Condensate 的工時 1,000 小時為何只是本文第 12.4.4(3)節所述及 2000 小時的一半？
12-029	核四所有人員(包括員工及包商)在進入電廠後均要換下私人衣服，改穿工作服？

編號	內　　　容
12-030	第 12.5.1.2(5)節電廠緊急計畫是由 HP 負責嗎？本節應與現行的台電組織分工一致。
12-031	請說明如何考量一部機組完工試運轉時，造成另一部建造中機組之施工人員劑量情形？
12-032	請說明第 12.2-9 頁大氣擴散因數(χ/Q)及沈積因數(D/Q)為 $2\times10^{-6}s/m^3$ 及 $4\times10^{-8}m^2$ 之合理性。
12-033	請說明表 12.2-23 中 Dilution Factor=10 之合理性。
12-034	第 12.2-10 頁 Reg. Guide 1.109 之 Dose Conversion Factors 與現行游離輻射防護安全標準是否一致？不一致處應予更新修正。
12-035	請說明放射性氣、液體排放造成廠外民眾劑量之計算模式及方法？
12-036	10CFR20 中有劃分超過 500 rad/hr 的輻射區，為何第 12.3.1.3 節沒有劃分出該種輻射區？
12-037	第 12.3.3 節應說明通風系統設計對於人員防護效果。
12-038	第 12.4 節劑量評估過程中，較 BWR 的減低劑量率的計算是否根據計算或經驗值？有否驗證？
12-039	龍門計畫之 PSAR 應考量除役之 ALARA，俾便確保整體的 ALARA 設計需求。
12-040	請說明 RWCU 系統之 filter/ demineralizers 所使用之樹脂，是否將採取再生式之設計。
12-041	請說明工作人員輻射劑量紀錄之保存方法及保存期限。
12-042	輻射工作人員在正常狀況下及異常事故後，如何實施體檢以保障其健康。
12-043	請說明劑量評估結果是否符合核能電廠環境輻射劑量設計規範之相關規定。
12-044	請說明 Health Physics Program 與台電現行作業之異同。
12-045	第 12.5-2 頁第三段似多餘，請說明之。
12-046	表 12.2-10，表 12.2-12 及表 12.2-17 之內容與 GE 公司 ABWR SSAR 之內容不同，請說明其原因。
12-047	PSAR 內容多處筆誤之處，請修正之。

第十三章　運轉管理

編號	內　　　　容
13-01	台電公司與 NSSS、BOP、A/E 之運作方式、管理與審查之規劃
13-02-1	說明石偉公司之組織狀況，有關核能審查經驗、人力配置及品質與品保方案執行經驗
13-02-2	建議增加一節：Plant Recrods
13-03-1	圖 13.1-2 組織與分工並未顯示 GE 及 stone & Webster 之地位及其對應之負責單位，宜澄清。
13-03-2	保安計劃應包括國內之勞工安全衛生法及消防法要求

第十四章　初始試驗計畫

編號	內　　　　容
14-01	施工後測試承諾，僅作原則性說明，請補充。
14-02	如何確保 TD 及相關人員具有足夠執行測試之能力及設計相關問題之處理。
14-03	測試方案和項目說明太簡略請補充
14-04	請補充如何符合 SRP14.2.II.5 要求
14-05	何時提出起動行政管理手冊？
14-06	請說明 MHI 之汽機起動測試與反應器界面
14-07	核四起動過程與現有廠之差異，以及控制棒操作方式是否不同？
14-08	汽機跳脫測試如何採用外插法？
14-09	SCRRI 是否可單獨執行，以及 RWM 和 MRBM 何以未包括阻棒功能測試？
14-10	何以 RFC 系統測試未包括 FTDC？
14-11	ESFAAS 系統測試是否包括反應時間測試？

第十五章　事故分析

編號	內　　容
15-01	FWCS 對 LFWH 的防範功能
15-02	SSAR 與 PSAR 在 LOCA 劑量分析之差異說明
15-03	LOCA 分析的劑量評估結果說明
15-04(1)	PSAR 暫態分析所用之參數
15-04(2)	PSAR 分析用 GEMINI option A 之說明
15-05(1)	T/B 跳脫時 Rx 跳機的 150ms 延遲說明
15-05(2)	表 15.1.7 的跳脫延遲時間說明
15-06	RWE 的事件分類
15-07	一次圍阻體小穿越管破裂之漏水量
15-08(1)	廢料廠房的耐震設計
15-08(2)	FHA 事件之能量分析
15-08(3)	FHA 事件之 both impacts 說明
15-09	說明 ODYNM 程式與 ODYNA 之差別
15-10	說明除 FWCS 外之 LFWH 之保護措施
15-11	說明何以在 HPCF 動作下，功率會下降
15-12	說明 LPZ 上甲狀腺劑量為 1.96 Sv 之問題
15-13	LOCA 分析的劑量結果
15-14	SSAR 與 PSAR 在 LOCA 分析之甲狀腺劑量差別
15-15	燃料束錯置之分析方法說明
15-16	燃料束誤旋轉之分析方法說明
15-17	說明劑量分析使用的各項參數
15-18	說明 PSAR 與 SSAR 在表 15.0-2 的差別
15-19	ARTS 與 APRM setdown 是否用於核四廠
15-20(1)	核四 SLMCPR 何以高達 1.09

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15-20(2)	何以 GE12 必須用 TVAPS 方法分析
15-20(3)	說明 MCPR 計算之統計校正因數
15-20(4)	說明何以△CPR 計算在 EOEC 最保守
15-20(5)	在 LRWB 分析中之 Rx 跳脫信號
15-20(6)	棄載且一個 TCV 失效之分析
15-20(7)	說明 combined steam flow limiter
15-21(1)	說明 MSIV leakage rate 值
15-21(2)	說明表 15.6-8 中各項疑點
15-22	說明得到表 15.6-9～12 的細節
15-23(1)	LOCA 評估的外釋途徑
15-23(2)	乾井集水池的外釋可能性
15-24	1.5.1.6 之公式說明
15-25	說明喪失冷凝器真空事件之細節
15-26(1)	表 15.1-7 的內容疑點說明
15-26(2)	RIP 跳脫事件時的回流現象說明
15-26(3)	ATWS 事件時的 permissive 信號說明
15-26(4)	說明 rapid core flow coastdown 之設定點
15-27(1)	ATWS 事件分析的 SLCS pump 數
15-27(2)	SLCS 動作時的時間延遲說明
15-27(3)	說明 ATWS 之事件序列
15-28(1)	FHA 事件之 failed rod 數量
15-28(2)	Failed rod 之燃耗假設是否保守
15-28(3)	Cask 中燃料數目是否保守
15-29(1)	RHR 停機冷卻誤開啟之頻率
15-29(2)	RIP 葉片斷落之衝擊能量

編號	內　　　　容
15-29(3)	何以圖 15A-51 未顯示 Rx 跳脫信號
15-29(4)	說明地震引發之事件
15-30(1)	說明控制棒插入時間之差異
15-30(2)	說明控制棒插入時間之保守度
15-30(3)	GE12 的 scram reactivity curve
15-31(1)	說明 15.3.1.5.2 之 special criteria
15-31(2)	Special criteria 是否可用在 GE12
15-31(3)	Special criteria 之燃耗要求
15-32(1)	何以沒有 LOOP 事件之分析
15-32(2)	在 LOOP 下的 Rx 跳脫信號
15-33(1)	正常運轉下時 FW 泵數目
15-33(2)	FWCF 事件說明
15-34(1)	核四自然循環流量相關問題
15-34(2)	ATWS 邏輯的動作元件
15-35	LOCA 分析劑量時之地形因素
15-36(1)	RIP 全跳脫事件之急停信號說明
15-36(2)	功率小於 80%時之 RIP 全跳脫事件
15-36(3)	不同功率發生 RIP 全跳脫事件說明
15-37	核四 LPZ 計算說明
15-38	核四爐心最大水流

第十六章　運轉規範

編號	內　　　　　容
16-01	台電公司提送流體排放運轉規範澄清
16-02	澄清 MFLPD 及 La 之定義是否納入。
16-03	澄清 16.3.2 未涵蓋 LHGR 與 APRM 原因。
16-04	要求增加再起動需原能會批准之規定。
16-05	1.澄清 SR 3 4.6.2 週期是否恰當。 2.澄清起動 RIP 未驗証爐心與爐水溫差之原因。
16-06	1.16.3.4.3 未有前 4 小時洩漏率≤2gpm 規定之原因。 2.16.3.4.5 未將測量總洩漏率系統納入原因。
16-07	澄清 16.5 節未納入相關規定之原因。
16-08	澄清 16.3.4.7 及 16.3.4.8 之 Bases。
16-09	澄清 SRV 可用性之相關問題。
16-10	澄清 page 16.B.3.1-6 中 30 之文字意義。
16-11	澄清允許 8 根控制棒急停時間超過之基礎。
16-12	澄清 RCIS 為非安全系統之原因
16-13	要求說明 Wrost Control Rod Configuration 如何決定。
16-14	要求說明 5×5Array Configuration 之 Bases.
16-15	要求說明"Special Reload Sequence"如何決定。
16-16	澄清棄載 150ms 之旁邊閥延遲是否入 T.S 中。
16-17	澄清 Table 16.3.3.1.1-1 之疑問。
16-18	澄清單一電源失效是否會造成兩串 SSLC 不可用。
16-19	要求說明 ECW.MCR-HVAC，RB-HVAC 未列入 T.S 原因。
16-20	要求澄清 SR 3.3.1.1.6，SR 3.3.1.1.10，16.3.3.1.2，16.3.1.4 中相關問題。
16-21	要求澄清 ECCS 測試要求，Completion Time 等相關問題。

審查問題與答覆內容

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-10-001

PSAR Sections: SECTION 10.4.6

Question Date: December 2, 1997

PSAR Question:

There is no increase oxygen equipment in Condensate Polishing System, how to control the resolved oxygen within the regulated value to prevent Corrosion/Erosion.

PSAR Response:

Dissolved oxygen levels are controlled by the deaeration function of the main condenser to levels below that required for the condensate polishing effluent as developed in the Plant Working Fluids Document and described below:

Fluid	Operating Target	Design Limit	Maximum Value
Condensate Influent (unit : ppb as O_2)	10	20	50
Condensate Effluent (unit : ppb as O_2)			
min	15	15	15
max	30	50	500

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

RESPONSES TO ROC-AEC's PSAR QUESTION

Track Number: **10-002**

PSAR Sections: **Chapter.10.2**

Question Date: **December 3, 1997**

PSAR Question:

1. Turbine generator is a high energy flying object, please specify the safety design criteria of a turbine generator. Please also provide supplementary explanation of the material(s) and test methods of turbine generator blades.
2. In the design of the Turbine Building , does S&W consider the enough lay down space to accommodate the disassembled parts from the overhaul?

PSAR Response:

1. Turbine orientation and placement within the turbine generator building is designed such that any plane perpendicular to the turbine generator axis shall not intersect with the primary containment structure. The probability of missile generation is less than 10^{-4} per reactor year. In addition, the probability of unacceptable damage to safety-related system and components due to turbine missile will be less than 10^{-7} per year.

   To prevent the occurrence of turbine missile accident, the following items are to be applied to the rotors and blades for Lungmen Project.

   <u>(1). Enhancement of quality control at manufacturing stage</u>

   <u>Improved manufacturing method.</u>

   At the manufacturing stage of rotors and blades, flaw and/or impurity of the material will be controlled to reduce thoroughly. The flaw and impurity has a possibility to cause the brittle fracture of rotors and blades. The improved manufacturing methods, such as smelting with ladle furnace and vacuum carbon degassing, are applied to raise the toughness of rotor materials for Lungmen Project.

   <u>Inspection, examination and test</u>

   For rotors and rotating blades, non-destructive inspection and/or programs are applied to confirm the accordance with design requirement.

   Rotor : Chemical composition check
   Mechanical properties test
   Non-destructive examination(UT,MT)
   Dimensional inspection
   High speed balance

   Blade :Chemical composition check

Mechanical properties test
Non-destructive examination(UT,MT)
Natural frequency measurement (for tuned blades)

(2). Redundant protective device

Overspeed trip system consists of the redundant devices, i.e. mechanical and electrical.

Surveillance tests for protective devices are to be conducted to maintain the wellness of the devices.

Steam inlet valves are provided as redundant in steam inlet line to prevent turbine overspeed by unclose of the valve.

In addition, the materials of turbine blades are 12% chromium steel and NiCrCu (Nickel Chromium Copper) steel.

2. The Overhaul Dismantling and Laydown plan is depicted on drawing 06888-1U72-M1020, which was issued with the Turbine Building General Arrangement drawings. All components are anticipated to be stored at the Operating deck elevation. The components noted on this drawing have been coordinated with the MHI turbine design. The turbine radiation cover segments are to be stored on the MSR and the Feed Pump compartment roofs.

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

RESPONSES TO ROC-AEC's PSAR QUESTION

Track Number: 10-003

PSAR Sections: 10.2

Question Date: December 9, 1997

PSAR Question:

1. Please provide closing and opening time of Main Stop Valves and Relief Valves respectively.
2. Does the turbine material meet the requirements of ASME code section III, NB-2500?
3. Are the Japanese Industrial Standards on Hi-Temp test equivalent to ASTM A-307 and E-208?
4. How is Lungmen T/G set designed to prevent the resonance frequency issue between turbine and main generator?
5. ASME Code Section XI requirements are followed for turbine in-service inspection. Which cod or standards is followed for the inspection classification? Is Volumetric Exam (e.g. ET.UT etc.) technique used for the inspection?

PSAR Response:

1. Under operating condition, Main Stop Valves close only in case of the valve test and fast closure for protection. For valve test, time to open or to close MSV is designed to satisfy the reactor requirement. For fast closure, it is also designed to satisfy Figure 10.2-1 in PSAR for Lungmen Project.
2. The components of turbine and their accessories are contracted to be in principle based on JIS (Japan industrial Standard) Codes and Manufactures Practices. Therefore the materials of turbine are not produced in accordance with ASME Code, NB-2500.
3. Standard mechanical testing methods designated in ASTM A 370 are described in JIS Cords Z 2241 to 2243, 2245 and 2248 respectively.

   Drop-weight test method to determine nil-ductility transition temperature, designated in ASTM E 208, are not applied to turbine materials, as the welded portions are not required to possess fracture toughness.
4. Turbine-generator shafting assembly is designed not to have the torsional natural frequencies within the ranges of 57 Hz to 63 Hz and 114 Hz to 126 Hz, so that no

torsional resonance between turbine-generator set and electrical disturbance force occurs.

5. Volumetric examinations are not required for turbine components in in-service inspection unless recommended by the turbine manufacturer.

RESPONSES TO ROC-AEC's PSAR QUESTION

Track Number: 10-004

PSAR Sections: Section 10.3.6

Question Date: December 3, 1997

PSAR Question:

According to the PSAR, low alloy steel was used for the contour nozzle of main steam line as well as for the nozzle on main feedwater pipe (between RPV and outside MSIV). It is different from the carbon steel stated in GESSAR. Please clarify how to prrevent stress buildup and cracking from happening due to welding after plant operation.

Response :

The Lungmen PSAR more clearly describes the actual design which is the same as what was used for Kashiwazaki-Kariwa 6/7. The main steam nozzles and the feedwater nozzles are low alloy steel as are all large nozzles welded to the reactor vessel shell. All BWRs use low alloy steel for large nozzles, and in many cases, these are welded to carbon steel pipes(e.g. Chinshan and Kuo Sheng main steam and feedwater nozzles). For Lungmen these nozzles will be made from SA-508, Class 3 forgings(earlier BWRs used SA-508, Class 2, also a low alloy steel).

The main steam nozzles has an extension welded to it that, in the original ABWR design concept was allowed to be either low alloy steel or carbon steel. However, when the detailed design was done for K 6/7, both reactor vessel manufactures elected to use low alloy and the same design is being applied to Lungmen. Consequently, the steam nozzles extensions will be SA-508, Class 3. The nozzle itself is post weld heat treated with the reactor vessel, the weld joint between the nozzle and extension is locally post weld heat treated after attachment to the nozzle, and the same is true for the weld joint between the nozzle extension and the pipe. Consequently all the low alloy steel joint are subjected to at least one post weld heat treatment as sepcified by ASME NE-4000. This heat treatment provides both stress relief and tempering of the as-welded microstructure so there is no concern for weld cracking. Further, there is no concern for in-service cracking of the joint between the low alloy extension and the carbon steel pipe. The moderate difference in strength does not produce a stress concentration and as noted above, most BWR reactor vessels have low alloy steel nozzles welded to carbon steel pipes. No incidences of in-service cracking have resulted from this configuration.

As with the main steam nozzle, the feedwater nozzle for Lungmen is low alloy steel(SA-508, Cl.3). The PSAR correctly clarifies that this nozzle is welded to a carbon steel safe end (SA-508, Cl. 1). Again the weld joint of the nozzle to the vessel shell and the weld joint between the nozzle and the carbon steel safe end are subjected to post weld heat treatment as required by ASME so there is no concern for weld cracking. The carbon steel safe end provides the transition to carbon steel piping. This configuration is identical to that used in K 6/7 and is generally common to most BWR feedwater nozzle/piping connections.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 10-005

PSAR Sections: Ch.10.4.3

PSAR Question:

What is the heating steam source for the Gland Steam System for both at normal operating and MSIV close condition? Is the steam source taken from the bottom up of the main steam when MSIV is closed? If the answer is yes, please verify whether the quantity from the bottom up of the main steam is enough. On the other hand, if the answer is no, the steam source should be obtained from auxiliary boiler. Then, please describe your design requirement and design base if any.

PSAR Response:

During normal plant operation, heating steam is supplied to the Gland Steam Evaporator directly from the Main Steam System piping downstream of the MSIVs.

Following MSIV closure, steam is provided from the Auxiliary Boilers directly to the Turbine Gland Sealing System, not to the heating steam side of the Gland Steam Evaporator. The Auxiliary Boiler will be maintained in standby during normal operation such that steam can be provided within 10 - 15 secs after receipt of an initiating signal.

No "bottled up" steam is needed; although any steam remaining downstream of the MSIVs will continue to flow to the TD Main Feedwater Pumps, Gland Steam Evaporator and other unisolated users as long as there is sufficient pressure downstream of the MSIVs to force a steam flow to these items.

RESPONSES TO ROC-AEC's PSAR QUESTION

Track Number: 10-006

PSAR Sections: 10.4.7.2.1

Question Date: December 3, 1997

PSAR Question:

Should the Hydrogen Water Chemistry System need to be added, is there sufficient space reserved for installing the associated equipment? Where the Hydrogen will be added? Please explain it.

PSAR Response:

Space is available in the Turbine Building on elevation -2500 or 12300 to accommodate future project decision relative to H_2 chemistry control.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **10-007**

PSAR Sections: **Section 10.3**

Question Date: **January 14, 1998**

PSAR Question:

What is the design leakage rate of MSIV? This information should be provided in this section. From Table 15.6-8, the leakage of MSIV from all lines is assumed to be 21.7 L/min which is roughly 0.3 m^3/h per line. However, in Chapter 16 page 166, surveillance requirement SR 3.6.1.3.13 requires to verify leakage rate through each MSIV ≤ 1 m^3/h when tested at ≥ 0.172 MPaG. The above two numbers do not seem to be consistent. Please clarify it. In addition, in Chapter 5 page 5.4-15 the air seat leakage test of MSIV is conducted using 0.28 MPaG pressure upstream. The test pressure is higher than that specified in Chapter 16 as mentioned above. Please explain the difference. The same page showed that maximum permissible leakage is 0.029 $cm^3/h/mm$ of nominal valve size. Please compare this number with the numbers mentioned above.

PSAR Response:

1. The design leakage limit of the MSIVs is 21.7 L/min for all four lines, as given in Chapter 15, Table 15.6-8. This limit is based upon the radiological release analysis for LOCAs inside containment in Chapter 15, Sec. 15.6.5. The following paragraph will be insert into Sec. 10.3.2.2:
   "The MSIVs are designed to limit the leakage to ≤ 21.7 L/min for all four lines, at a pressure corresponding to the calculated peak containment pressure for design basis accidents given in Table 6.2-1".
   In addition, the following paragraph will be added to Sec. 10.3.4, Inspection and Testing Requirements:
   "The MSIVs shall be tested to assure that leakage is ≤ the specified leakage limit stated in Sec. 10.3.2.2, in accordance with Chapter 16, Technical Specification surveillance testing SR 3.6.1.3.13".
2. The surveillance requirement SR 3.6.1.3.13 in Chapter 16, Technical Specification, currently states:
   "Verify leakage rate through each MSIV is ≤ 1 m^3/h when tested at $\geq$0.170MPaG".
   The requirement is based upon ABWR SSAR. The MSIV leakage limit shall be made consistent with the design leakage limit of 21.7 L/min (1.3 m^3/h) as stated in Item (1) above. Appendix J of 10 CFR 50 requires that local leak rate tests (Type C tests) be performed periodically. The test pressure for such air tests shall be P_a, the calculated peak containment internal pressure related to the design basis accident. The calculated peak containment internal pressure for

design basis accidents (LOCAs inside containment) is 268.7kPaG as given in Chapter 6, Table 6.2-1, Containment Parameters. Therefore, the test pressure for the MSIVs shall be ≥ 269kPaG. Chapter 16, Technical Specification, SR 3.6.1.3.13, will be revised as follows:

"Verify leakage rate through each MSIV is ≤ 0.32 m^3/h when tested at a pressure ≥ 269kPaG".

3. In Chapter 5, Sec. 5.4.5.3, an air seat leakage test is specified to be conducted using a test pressure of 0.28 MPaG. The maximum permissible leakage is also stated as 0.029 $cm^3/h/mm$ of nominal valve size. These values are based upon the ABWR SSAR. Both the test pressure and the permissible leakage in Item (4) on page 5.4-14 will be revised as follows:

   "Leakage is measured with the valve seated. The specified maximum seat leakage, using cold water at design pressure, is 56 cm^3/h. In addition, an air seat leakage test is conducted using 269 kPaG pressure upstream. Maximum permissible leakage is 0.079 m^3/h".

   Testing for a smaller leakge limit in the shop is a conservative measure and enhances the probability that the leakage rate limit during periodic in-service surveillance test will be met.

   In Sec. 5.4.5.4, Items (2) and (4), all references to a test range of 0.14 to 0.21 MPaG will be changed to "*269-310 kPaG*" to be consistent with the test pressure for Lungmen. The pressure of 269 kPaG and 310 kPaG are the P_a and containment design pressure, respectively, as stated in Table 6.2-1.

ROCAEC Review Comments :

1. It can be seen from the response that the leakage rates for the MSIV are planned to be specified at 0.32 m^3/h.......per valve which is equivalent to 21.7 L/h.... for all four lines. But from PSAR Appendix AJ page AJ.3-22, operational experiences showed that 50% of valves can not satisfy 0.33 m^3/h technical specification requirement. Unless it can be shown that MSIV at Lungmen has a better design, it is not acceptable to specify such a leakage rate.
2. It can be seen from the response that in the LOCA analysis, the above MSIV leakage rate was used to calculate the radiation leakage. If the above leakage rate can not be proven to be appropriate, then the LOCA analysis should be re-evaluated.

Further Clarification:

1. The MSIV Leakage rate for Lungmen was reduced from the ABWR Certification value to a value used in all BWRs prior to 1991. Such a value is more demanding from a maintenance perspective but has historically been achieved. Lungmen is expected to have better leak rate test results than earlier plants due to demonstrated improvements in the MSIV design as described below.

In the US, the NRC has approved utilities that operate BWRs to adopt a less restrictive leak rate for the MSIVs. Instead of 0.32 m^3/h per valve, the NRC allows a leak rate of 2.83 m^3/h for each steam line. However, if a MSIV fails this leak rate, the valve has to be restored to a leak rate of 0.32 m^3/h or less before the plant is restarted

The MSIVs that GE is supplying to Lungmen do have an improved design. The valves have incorporated design improvements such as:

A. A longer valve stroke (13 3/4 inch compared to 10 inch stroke), With the longer valve stroke for Lungmen, the MSIV flow configuration is within the vendor tested flow range. Thus, the vendor prediction for the pressure drop of the MSIV is more credible.

B. Qualified actuators and stronger/larger structural components such as the yoke rods,

C. Longer end to end length i.e. 72 inches long weld to weld length in the steam line (compared with 60 inch length for MSIVs in Browns Ferry),

D. Live loaded packing,. This feature develops less stem friction than the original packing configuration and keeps the friction constant over the life of the packing. This feature, when used with the cover improvement, eliminates the packing leakoff.

E. Poppet backseated cover. This feature was developed to eliminate the rib wear caused by poppet movement while it is in the open position. This feature locks the poppet to the valvecover while the valve is in the open position.

F. Antirotation stem and poppet. This arrangement was developed to prevent rotation of the poppet or stem, and also assures there will be repeatability of seating surface contact.

G. Improved stem guidance system. This feature helps eliminate stem breakage and provides increased stem guidance. A second stem bushing was added, and the stem diameter is a half inch larger than the diameter which passes through the original bushing. This larger stem diameter greatly increases the rigidity of the stem where the greatest forces are felt should poppet movement cause stem side loading.

H. A re-designed poppet nose and higher lift poppet are the result of the work that the vendor has done for the ABWR when it was under study earlier in Japan. This was developed to compensate for the seat friction caused by the film build up on the stellite body and poppet seats, This feature guides the poppet into the valve body seat.

I. A floating pilot poppet for the best seating and pressure drop improvement. The floating pilot poppet has been incorporated because it has been the vendor observation that the valves have a better LLRT results than those that do not have (i.e. typically 7 or more of the MSIVs passed the LLRT compared with typically 6 or more of the MSIVs failed the LLRT). This is because the floating pilot poppet will allow for slight mismatch of seating surfaces as the pilot poppet will tend to find its own best seating position.

J. One piece forged stem. This feature will not allow the poppet from separating from the stem.

All these features incorporated in the valve design have been proven in domestic operation asvalve improvement modification over the years (such as in Brown Ferry Unit 1, 2 & 3) , The leak tightness of the valves is much better.

2, The LOCA analysis was indeed based upon the MSIV leakage rate described above. Accordingly, Lungmen maintenance practices will be developed to assure that such leakage limits can be demonstrated during leak testing. This improved design of the MSIVs to be used in Lungmen will assist in making this achievable.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 10-008

PSAR Sections: 10.4.5.2.2

Question Date: Nov. 20, 1997

PSAR Question:

Some of the information in Table 10.4-3 on page 10.4-28, regarding the flow capacity of the circulation pumps, was not specified, please provided the detailed information.

Response:

Table 10.4-3 should be revised as follows to include the previously missing detailed information:

Circulating Water Pumps		
	Number of pumps	6
	Pump type	Vertical, wet pit
	Unit flow capacity, m^3/hr	59,000
	Driver type	Fixed speed motor
System Features		
	Pump discharge valve & actuator	Butterfly, motor
	Condenser isolation valve & actuator	Butterfly, motor
	Condenser tube cleaning equipment	Ball type in conjunction with debris filter upstream of condenser
	Debris Filters	Mesh size 8 mm maximum
	Traveling water screens and screenwash pumps	Dual flow type with 2-100% screenwash pumps with downstream strainers
	Vacuum priming system	2-100% motor driven

RESPONSES TO ROC-AEC's PSAR QUESTION

Track Number: 10-009

PSAR Sections: 10.4.1.1.2

Question Date: Nov. 20, 1997

PSAR Question:

The main condenser is designed to sustain the turbine by pass system for 110% main steam flow rate, the capacity and heat transfer capability are relatively greater, is it the Proven Design? Is there any operation experience for the similar type of the main condenser experience in any power plants outside of this country?

Response:

Although not conventionally provided in U.S. plants, large bypass designs are often included in plants (especially European and South African fossil units) connected to small electrical grids. This design philosophy was developed in Europe and is in use in many countries with relatively small grids. Large bypasses are used to protect the grid from large load swings and allow the plants to restart quickly.

The main condenser component part design features will not be affected by the capacity of the bypass except for component part sizes. The capacity of the main condenser has been specified such that trip will not occur even with the highest inlet circulating water temperature and full steam bypass flow with the setpoints provided in Section 10.4 of the PSAR. Steam bypass valves and control features are the limiting features in a design of this nature.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 10-010

PSAR Sections: 10.4.1.2.1

Question Date: Nov. 20, 1997

PSAR Question:

The three condenser shells are cross-connected to equalize pressure, but GE SSAR described that condenser shells have portions of low-pressure, intermediate-pressure and high-pressure. Is there any difference for the aspects of design and function? Please clarify.

Response:

The GE SSAR essentially reflects a forecast that a multipressure condenser arrangement would be utilized for a six flow turbine since it would likely be a condenser designed for a closed-cycle cooling system where the related cooling water costs would be relatively expensive. In contrast, the Lungmen condenser design was selected to be compatible with an open-cycle cooling system which employs large quantities of relatively inexpensive cooling water.

The Lungmen design of three once-through condenser shells in parallel is mechanically simpler than a multipressure arrangement and allows more flexible operation since the six condenser tube bundles are independent of each other. On the other hand, a multipressure condenser configuration requires full height compartmentation to separate each turbine back end, a means of condensate reheating that among other features, generally necessitates a constrained location for the feed pump turbine exhaust. It must also have false decks under the tube bundles to isolate the condenser hotwell and will have interdependent tube bundles during operation. The multipressure design requires a sequential circulating water path under each of the three separate low pressure turbines that causes the turbine exhaust pressures to be continuously raised such that they are respectively designated as the low, intermediate and high pressure compartments. In this manner, the multipressure design effectively utilizes the circulating water flows that are limited by the cost of the accompanying closed-cycle cooling system, but this condenser is more expensive than the type selected for Lungmen and creates a higher average turbine backpressure. That is, everything else being equal, besides its operating flexibility that permits only one sixth of the condenser to be removed from service for maintenance, the Lungmen condenser configuration is capable of lower turbine exhaust pressures and will always produce a higher unit generation; in addition, the Lungmen design will produce minimal condensate subcooling and lower levels of dissolved oxygen in the condensate.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **10-011**

PSAR Sections: **10.4.1.1.2**

Question Date: **Nov. 20, 1997**

PSAR Question:

In "Power Generation Design Basis Four" of the main condenser, it was mentioned that under the normal operation condition, the main condenser hotwell will provide a condensate storage volume and surge capacity for at least 4 minutes or under the full load operation for at least 2 minutes. In addition, the design may effectively delay condensate water flow for at least 2 minutes for decay of N-16 radiation.

Compare with Kuoseng NPS, Kuoseng plant can maintain up to 5 minutes under the full load capacity. Also the plant capable to delay the condensate water flow at least 5 minutes in order for radiation decay.

Please specify that if the condenser hotwell capacity design of the Lungmen NPS is proper? What is the justification of the design consideration for the Lungmen NPS?

Response:

Conventional designs include a main condenser hotwell level equivalent to 3-5 minute Valves Wide Open (VWO) feedwater flow. The Lungmen design of 4 minutes is based on the TPC Bid Specification requirements of Appendix A Chapter 2 Section 4.2 and is within the conventional design criteria.

From a radiological perspective, the hotwell holdup time must be sufficient that the N-16 activity leaving the hotwell is (a) small compared to the halogen/fission products/corrosion product activity in the condensate, and (b) the unshielded dose rate from a large pipe carrying condensate allows continuous personnel access. Based on a half life of N-16 (the leading source of radiation) of approximately 7.13 seconds and the expected condensate concentrations, approximately 146 seconds is required for N-16 to decay to acceptable levels. The contact dose rate from a 10 meter long 650 m pipe carrying condensate with the expected N-16 levels is expected to allow continuous personnel occupation. Therefore, radiation decay will be essentially complete within approximately 2.5 minutes. It is, therefore, concluded that a holdup time of 4 minutes is adequate for radiation decay and the radiological requirements are met.

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

RESPONSES TO ROC-AEC's PSAR QUESTION

Track Number: 10-012

PSAR Sections: 10.3.2.1

Question Date: November 20, 1997

PSAR Question:

In section 10.3.2.1, the following statement is given: "The steamline drains, except through Control Building, maintain a continuous downward slope from the steam system low points to the orifice located near the condenser...". What is the main purpose for the continuous downward design? Is the prevention of water hammer part of the major concern? For the portion of control building, what provision has been made to fulfill the purpose of continuous downward design?

PSAR Response:

Prevention of water hammer due to water collecting in the drain line is one of the reasons for maintaining a continuous downward slope in the line. The steamlines and the bypass/drain line are arranged to provide gravity-driven draining.

The steamlines and the bypass/drain line are arranged to maintain a continuous downward slope in the direction of flow to the low point in the Reactor Building steam tunnel. Then, the lines slope up to reach the high point in the steam tunnel on top of the Control Building.

In the Control Building, the steam lines and the bypass/drain line are sloped down in the direction of flow from the high point in the steam tunnel on top of the Control Building to the piping interface one meter outside of the Control Building.

Section 10.3.2.1 will be revised to incorporate the following statement:

" ----- The steamline drains, [<u>including drains</u>] through Control Building, maintain a continuous downward slope [<u>in the direction of flow to</u>] the steam system low points [<u>in the Reactor Building steam tunnel and then slope up to reach the high point in the steam tunnel on top of the Control Building. The purging process in this part of the line makes use of the bypass/drain line air-operated orifice bypass valves with a valve opening time of greater than and equal to 30 seconds, permitting a steadily increasing bypass steam inflow. With the in-line multi-stage pressure reducing orifices, and associated control actions with the drain line valves, this will assure that the water collected in the line will be adequately drained to the main condenser and with no adverse water hammer effects.</u>] [<u>From this high point in the Control Building, the lines slope downward in the direction of flow to the NI-BOP piping interface one meter outside of the Control Building.</u>] The drain line from the -------- to the radwaste system."

ROCAEC Review Comments:

Not accepted for the moment.

1. In the Main Steam Tunnel, the main steam line and the main steam bypass line, etc. have a low point and a high point in the control

building and there is a need of a drain for the low point otherwise water hammer phenomenon will occur. Please explain (1) why the steam line goes down first and then goes up before it finally goes down in the control building; (2) will there be maintenance problems if drain line is installed in the main steam tunnel ?

2. During load rejection and turbine trip, the opening of bypass valve on the main steam bypass line will determine if the reactor will scram so its opening speed must be very fast. However, in the response it was stated that the bypass valve opening speed is greater than 30 seconds, which seems to imply that the opening should not be too fast to avoid water hammer phenomenon. Please clarify if there is any contradiction.

Further Clarifications:

1.

(1) In the Lungmen plant arrangement, the Reactor/Control Building is at a lower elevation than the Turbine Building. In order to provide gravity-drive draining, the steam lines and the main steam line bypass drain line are arranged to maintain a continuous downward slope in the direction of flow to the low point in the Reactor Building steam tunnel. Then, the lines slope up to reach the high point in the steam tunnel on top of the Control Building. Therefore, it is not possible to accommodate gravity-driven draining to the main condenser in the bypass drain line alone. Draining by differential pressure will also be required. At the low point in the steam tunnel, drains are provided for maintenance purposes only.

The purging process in the main steam line bypass drain line has been included in the system design to assure that the water collected in the line will be adequately purged and drained to the main condenser and with no adverse water hammer effect.

(2) Low point drains to the LCW are provided in the main steam line bypass drain line in the main steam tunnel for draining of the main steam lines during an outage. These drains are manually operated and closed prior to a plant startup and do not require maintenance during normal plant operation, and no maintenance problems are anticipated. These drain valves are standard design features for BWRs.

2. The drain bypass valves and the turbine bypass valves are designed for different functions. During normal plant operating condition, the main steam line bypass drain valves will be closed when the reactor is at > 40 % NBR condition, so that steam loss to the main condenser will be minimized. This drain line to the main condenser will only be open during plant startup and when the reactor is at ≤ 40% NBR condition. The bypass drain valves are small MOV that only open to pass drain flow.

As for the turbine bypass valves, they are designed to open quickly after a load rejection or turbine trip events. The turbine bypass valves are hydraulically operated valves that open in 150 milliseconds and have a capacity of 110% NBR.
There is an apparent misunderstanding of the usage of two different terms - the drain bypass valves and the turbine bypass valves.

No change to the PSAR is required.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 10-013

PASR Section: 10.2.3

Question Date: Nov. 21, 1997

PSAR Question:

In section 10.2.3.1 Materials Selection, it was stated that actual levels of FATT and Charpy V-notch energy will be obtained through precise destructive tests of actual samples from each turbine rotor. Please explain how the samples are going to be obtained and how to maintain the integrity of the rotors?

Response:

The sample materials used for Charpy test are taken from the surplus portions adjacent to the outer surface of turbine rotors as the below figures. Therefore, the integrity of the rotor is maintained.
(The figure is manually cut and paste from file:"10-013 MHI resp.tiff'

RESPONSES TO ROC-AEC's PSAR QUESTION

Track Number: 10-014

PSAR Sections: 10.1

Question Date: November 20,1997

PSAR Question:

1. In Table 10.1-1 of page 10.1-6,please provide information on the Rated motor power (kW) for the Main pumps of the Reactor Feedwater Pumps.

Response:

1. The rated motor power (kW) for the Reactor Start-up Feedwater Pump is a nominal 4500 kW as shown below in the Lungmen PSAR Table 10.1.1

Table 10.1.1
Summary of Important Design Features and Performance Characteristic of the Steam and Power Conversion System
Reactor Feedwater Pumps

Number of pumps	3 turbine driven / 1 motor driven startup
Pump type	Horizontal centrifugal
Driver type	3 turbine driven / 1 motor drive startup
Design conditions:	
Main pumps:	
Normal flow, m^3/hr	2700 m^3/hr
Total head, meters	700 m
Rated motor power, kW:	N/A (Turbine Driven)
Startup pump:	
Normal flow, m^3/hr	1600 m^3/hr
Total head, meters	700 m
Rated motor power, kW:	4500 kW

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 10-015

PSAR Sections: 10.4.5.3

Question Date: Mar. 24 1998

PSAR Question:

1. Regarding to CWS Flooding Analysis, in SSAR was mentioned that ".... for conservatism, it assumes that one system isolation valve does not fully close.". Why the statement is eliminated on PSAR Section 10.4.5.3? please justify. Is this assumption not suitable for Lungmen Nuclear Power Project?

Response:

The statement "...for conservatism it assumes that one isolation valve does not fully close" was intentionally deleted form the Lungmen PSAR as it was not considered conservative with the Lungmen Nuclear Project because the valve that services the failed expansion joint is assumed to close upon CWP trip. If any other condenser inlet valve does not completely close it has no bearing on the flooding analysis. In any event the level instrument in the condenser pit will trip all CWP's to preclude flooding of the Turbine Building. After the CWP's have immediately tripped only water that is in the affected condenser waterbox may discharge through the failed expansion joint.

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

RESPONSES TO ROC-AEC's PSAR QUESTION

編號(Track Number)： N/A

問題章節(PSAR Section)： 10.4.5

初提日期(Question Date)： 1998.08.21

問題內容(PSAR Question)：

請補充說明循環水出口溫度及最大溫升，以及距離排放口500公尺之溫升並評估是否符合環保署所公佈放流水不得超出4℃溫升之規定？

問題答覆(Responses)：

(1) 一、龍門核能電廠循環水系統之設計，將使通過冷凝器後之冷卻水溫升不會超過7℃，且在離排放口500公尺處之溫升不會超過4℃，完全符合台電於環境影響評估報告中之承諾，也符合環保署放流水溫升規定。

(2) 二、將修改龍門初期安全分析報告第10.4.5節如下：

"10.4.5 Circulating Water System

The Circulating Water System (CWS) provides cooling water for removal of the power cycle waste heat from the main condensers and transfers this heat to the power cycle heat sink. The temperature rise of cooling water passing through the condenser will be no greater than 7℃ and the temperature rise at 500 meters away from discharge point will be no greater than 4℃ which comply with the commitment of Lungmen Environment Impact Assessment."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **R-11-01**

PSAR Sections: **11.5**

Question Date: **December 16, 1997**

PSAR Question:

PRM (Process Radiation Monitoring) system should have its own specific section for complete description which covers (1)PCS (Plant Computer System), (2)Radiation Monitors for Safety and Protection, (3)Radiation Monitors for Plant Operation, and (4)Trip Circuit; monitors which have their detectors and instruments at the same location and monitors which have their detectors at the field and instruments in the control room are all identified. All monitors' signals are transmitted to the PCS in the control room for storage with a fixed cycle time (usually 1 second). These signals can be used for real time display or long term history use (usually 1 day) by selecting one or several of them. PCS screen should have display similar to Fig. 11.5-1 (Location of Process Radiation Monitor) and Table 11.5-8 which shows instantaneous, monitored data.

Response:

The configuration of and the data to be displayed on the Main Control Room operator's screen will be determined using Human Factors Engineering during the detailed design. Radiation information that is considered necessary for safe, efficient and reliable operator performance during all phases of normal, abnormal events and accident conditions will be displayed. Section 11.5, 3rd paragraph, last sentence of the PSAR will be modified to state that:

"All non-safety-related radiation monitors that are contained in the Process and Effluent Radiation Monitoring System are on-line networked and are configured to continuously transmit information back to the Plant Computer System (PCS) via the Non-Essential Multiplexing System. The PCS is capable of displaying necessary information to the operator. All safety-related radiation monitors that are contained in the Process and Effluent Radiation Monitoring System are hardwired to class 1E Remote Multiplexing Units of the Multiplexing System (MUX). The MUX provides a redundant and distributed control and instrumentation data communications network. For the status monitoring, the MUX provides, via a 1E to non-1E gateway to the PCS, support for the status display and alarm of the radiation monitoring system."

The change to Section 11.5 will be made to the final revision of the PSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: R-11-02

PSAR Sections: 11.5

Question Date: December 16, 1997

PSAR Question:

Contents of Section 11.5.3 (Effluent Monitoring and Sampling) and 11.5.4 (Process Monitoring and Sampling) on Sampling are too rough (almost just one sentence). Only Tables 11.5-4, 11.5-5, 11.5-6 and 11.5-7 were presented and even those 4 Tables were not explained in text either. Sampling Description should include, besides the sampling goal, the location, methodology and equipment, etc.

Response:

Per the Standard Review Plan (SRP) for Chapter 11.5, information pertaining to the location of sampling points, sampling stations and related equipment need only to be defined in the FSAR as appropriate. Basic information was provided in the PSAR only to aid the reviewer as to the overall scope of the Process Radiation Monitoring System. The following text will be added to the end of Section 11.5.3.3:

"Tables 11.5-4 through 11.5-7 provide summary information concerning the frequency, analysis, sensitivity and purpose for both liquid and gaseous process and effluent extracted samples that are analyzed in the health physics laboratory."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **R-11-03**

PSAR Sections: **11.5**

Question Date: **December 16, 1997**

PSAR Question:

1. Tritium is a Beta-Emitter. This section does not include tritium monitoring or how its release rate can be evaluated. Please explain.

2. Second paragraph from the last of Section 11.5.2.2.4 reads "The ranges of channel measurement are....; and 3.7E-3 Bq/cc to 3.7E1 Bq/cc of tritium." Is it feasible to have on-line monitoring system for direct measurement of tritium ? How is the calibration done ?

Response:

1. Information pertaining to the monitoring of tritium in effluent paths is provided under the individual subsystems that should be monitored for this radioisotope. For example, in 11.5.2.2.4, which serves as the confluence for several radioactive paths, tritium monitoring is mentioned. Each effluent path that requires tritium monitoring will be provided with the capability to detect the appropriate regulatory limits.

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

2. It is feasible to have on-line monitoring for tritium, although it is not strictly required by regulatory guidelines. There are two basic approaches by which tritium monitoring can be accomplished. The first approach adds the capability to continuously extract a tritium sample through a desiccant cartridge for laboratory analysis. Tritium sampling, in this case, is accomplished by placing parallel desiccant collectors complete with isolation and throttling valves, on the exhaust side of the noble gas monitoring channel. The tritium sampler is installed in series with the noble gas sampler between the noble gas sampler and the inlet to the vacuum pump. This arrangement passes the sample stream through one of the sampler elements, while the other

is available for immediate change-out. This method allows sampling to proceed continuously without interruptions for desiccant removal. The second approach provides continuous real-time monitoring by placing an ionization chamber based monitor on the exhaust side of the noble gas channel. The tritium monitor can be placed in parallel with the main sample stream, and, using vacuum/flow regulation, a sample can be continuously withdrawn and measured. The actual equipment to be supplied for Lungmen NPS will balance the need for on-line versus periodic sampling on a technology and cost basis. Calibration procedures associated with an on-line device would be vendor specific and will not be known until the equipment is specified and procured.

Approach 1, described above, may have the technical concern of slow response time and the need to periodically change the filters. The only known technical problems with online monitoring utilizing Approach 2 is the potential for residual contamination inside the ionization chamber to effect the reading.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **R-11-04**

PSAR Sections: **11.5**

Question Date: **December 16, 1997**

PSAR Question:

Page 11.5-7, line 7 reads "The design basis is selected to initiate isolation of the MCR prior to exceeding 0.05 Sieverts whole body..." which complies with the current annual dose limit for ionizing radiation protection personnel. However, Lungmen will come on line quite a few years later and ICRP-60 will be adopted by many countries including ROC, so it's imperative that the design of radiation protection should have margins for future regulation changes. Please explain the approaches taken.

Response:

The radiation monitors for the Main Control Room Ventilation Intakes are intended to provide initiation signals to the proper air handling equipment to ensure adequate ventilation control for Main Control Room personnel under accident conditions. Upon recognition of an exceeded radiation level, trip signals are sent to the Main Control Room HVAC system to initiate the appropriate actions to protect the control room personnel from excessive radiation. The radiation monitors are designed to have adjustable trip setpoints that correspond to the recommended regulatory guidelines. As required by USNRC Regulatory Guide 1.105, equipment specific errors, location specific details and process errors must be included in the setpoint determination. These setpoints will not be determined until the FSAR stage.. The installation of these monitors complies with 10CFR50 Appendix A, General Design Criterion 19. The dose mentioned is consistent with that listed in Ionizing Radiation Protection Safety Standards for annual dose limit of an occupational exposure, i.e., 50 mSv (0.05 Sv) in one year.

The list of applicable Codes and Standards, for which Lungmen NPS radiation monitoring is to be designed, does not include future regulations, such as ICRP 60, since currently there is not a requirement to do so. Since the range for the Main Control Room (MCR) Ventilation Intake radiation monitors are from 1E-4 mSv/h to 1E0 mSv/h, and based on historical data from operating BWRs, the setpoint for these monitors is typically 1E-2mSv/h. Thus, it is anticipated that there will be sufficient margin on the range for the MCR Ventilation Intake radiation monitors so that the range will be sufficient to meet potential future reductions in radiation limits. If, however, the sensitivity requirement is dropped significantly beyond the capability of the installed radiation monitor, then the radiation monitor will obviously need to be replaced with an appropriate design compliant with the regulations at that time..

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **R-11-05**

PSAR Sections: **11.5**

Question Date: **December 16, 1997**

PSAR Question:

Page 11.5-23, Table 11.5-2, column 3 where listed values were new radioactivity unit Bq which is converted from old radioactivity unit Ci so the conversion factor of 3.7 was involved. ROC has adopted the new unit in the Ionizing Radiation Protection Safety Criteria. In the future, all monitoring instruments will be brand new, so it should not be just conversion from old units. Also, in the 5th column of the same Table, the estimated activities at various places were all 1.48 Bq/cc. Why are they identical ?

Response:

The specified dynamic range of the instrument, i.e., the capability of the instrument to measure over a minimum range in a stated engineering unit, does not preclude it from displaying in other engineering units. During the detailed design and procurement stage of the Process and Effluent Radiation Monitoring subsystems, human factors consideration will be given to the choice of units and how they relate to the process to which the radiation instrument is related and an appropriate display unit will be selected. In Column 3 of Table 11.5-2, the units will be changed to "Bq/m^3" where they are currently shown as "Bq/cc".

The value of 1.48 Bq/cc for these entries in column 5 were considered to be approximately equal based upon typical releases from these ventilation paths. Although actual Lungmen NPS activities may vary somewhat from these values, the use of these values has historically be shown to be sufficient for proper selection of the range for the radiation detectors.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **R-11-06**

PSAR Sections: **11.5**

Question Date: **December 16, 1997**

PSAR Question:

In page 11.5-25, Table 11.5-4, the 4th column has unit MBq/L which is not consistent with the unit Bq/cc used in other Tables. They should all be changed to unit Bq/m3 to be consistent with the unit used in Table 4 of the ROC Ionizing Radiation Protection Safety Criteria.

Response:

All entries in the fourth column of PSAR Table 11.5-4 will be changed from "Bq/L" to appropriate values expressed in "Bq/m^3".

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: R-11-07

PSAR Sections: 11.5

Question Date: December 16, 1997

PSAR Question:

In page 11.5-26, Table 11.5-5, 3rd column, it reads "Gross alpha & beta Tritium". Is it correct to list them in the same column ? Is it really alpha ?

Response:

The entry should restructured to be two separate lines, so that it reads as follows:
"Gross α and β
Tritium"

As regards the question "Is it really alpha", the response is "Yes".

The change will be made to the final revision of the PSAR as shown in the first sentence above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **R-11-08**

PSAR Sections: **11.5**

Question Date: **December 16, 1997**

PSAR Question:

In page 11.5-27, Table 11.5-6, 4th column, is the Gross alpha value too high ?

Response:

The following change will be made to PSAR Table 11.5-6, 4th column: "Gamma spectrum" will be corrected to "18.5 x 10^{-6}". The value for gross alpha is correct per Regulatory Guide 1.21.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: R-11-09

PSAR Sections: 11.5

Question Date: December 16, 1997

PSAR Question:

In page 11.5-28, Table 11.5-7, the text in columns 2, 3 &4 reads "As above" which is not very clear as to what it means and should be corrected. Also, in column 3, the symbol after "Gross alpha" is not clear as to what it represents ?

Response:

"As above" was meant to indicate that each sample point, i.e., Plant Stack Discharge, Gland Steam Condenser Exhaust Discharge and Radwaste Building Discharge, will have the same sample frequencies of weekly, monthly and quarterlyy as shown for the same items as shown for the Ventilation Stack Discharge. That is, for example, the "Plant Stack Discharge" will have the same analyses, with the same frequency, as shown for the "Ventilation Stack Discharge". The same analyses and frequencies also apply for the "Gland steam condenser exhaust discharge" and the "Radwaste Building Discharge".

The symbol after the "Gross Alpha" was meant to be "†", i.e., to indicate, via the footnote, that the sample is collected on a particulate filter. In the final revision of Table 11.5-7 of the PSAR, the symbol will be changed accordingly. In addition, in the final revision of the PSAR, in Table 11.5-7, "**" will be added to the entries indicated as "As above" and an additional footnote will be added stating "** Same sample frequencies of weekly, monthly and quarterly as above for the same item as shown for the Ventilation Stack Discharge."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **R-11-10**

PSAR Sections: **11.5**

Question Date: **December 16, 1997**

PSAR Question:

Please explain the differences between Lungmen and other BWRs in areas such as Source Terms, gaseous and liquid effluents. Also, please explain how the design bases for Lungmen were established.

(Addendum) Please provide a comparison table between Lungmen and typical BWR on Source terms, gaseous and liquid effluents to show the difference. If Lungmen adopts a better system than ABWR, please explain the advantages to the environment.

Response:

There are no significant differences between the Lungmen NPS source terms and those of previous designs. The differences which can be seen are minor and are due to specific differences in the components and operations of the Lungmen plant compared to other BWRs. This is to be expected since the plant involves the same basic components and operations as other BWRs. First, the basis for the source terms will be explained and then the differences in the design which are unique to the Lungmen NPS.

The source terms for the Lungmen NPS consist of two components, (1) the core sources, and (2) the fluid sources which are broken down into (a) the water sources and (b) the steam sources.

Firstly, the core source terms, which are not supplied in the PSAR except those needed for Design Basis Accidents but are given in the Radiation Data Book (to be supplied) provide an isotopic breakdown of the fission products in the core for an equilibrium BWR core in terms of Bq per MWt. This is a GE standard source for plants with burn-ups on the order of Lungmen NPS to be used for design basis evaluations and represents a reasonable yet conservative evaluation of the expected sources in a shutdown equilibrium core.

The water and steam sources are derived partly from historical precedent as well as from operating experience. The water and steam source terms are calculated using ANSI/ANS 18-1-1984 to evaluate relative isotopic ratios for those radionuclides normally found in BWR water/steam. The results of the ANS 18-1 calculation are then modified as follows.

- For the 13 long lived noble gas isotopes, a ratio is determined and the releases modified by this ratio such that the total release rate of those 13 isotopes equals 3,700 MBq/sec (evaluated for a time 30 minutes downstream from the vessel exit nozzle). The remaining nobles gas isotopes are also multiplied by the same ratio to complete the noble gas source terms.
- In a similar fashion a ratio is calculated for Iodine-131 using a core release rate of 25.9 MBq/sec of Iodine-131 and all the other isotopes except H-3 (Tritium) and the activation products (Nitrogen-16 and others) are ratioed up by the number found based upon Iodine-131.

These values of 3,700 MBq/sec (100,000 μCi/sec) for noble gases and 25.9 MBq/sec (700 μCi/sec) for Iodine-131 were developed in the study reported in NEDO-10871 ("Technical Derivation of BWR 1971 Design Basis Radioactive Material Source Terms") and has since been know as the "71 Source Term". All BWRs since then have used this as the design basis primarily to protect the utility investment. In the late 60s, BWRs were operating at values just below these source terms. Over the two and one-half decades since then, fuel performance and system improvements have reduced operating values such that the Lungmen NPS should show offgas values of 5-10% of the design noble gas rates and water/steam concentrations 1-5% of the design values. However, to protect the utility from off normal conditions such as might occur if a batch of poor performing fuel were to be introduced, the design basis is set to these higher limits such that the reactor could operate for extended periods under it current license until a planned outage could be instituted and the condition rectified.

For Tritium and activation products, standard source terms from the above ANSI document are used except for the case when Hydrogen Water

Chemistry in which case a source term of six times the total activation source term of 1.85MBq/gm steam (50 μCi/gm steam) is used. This increased activation source term represents a bounding value for the ranges of hydrogen injection seen in current use.

The Lungmen NPS varies from most BWRs by utilizing cascaded condenser drains which prevents the recirculation of contaminates in the steam from being reintroduced into the reactor feedwater which is the case with most BWRs including the ABWR Certified Design. In addition, the Lungmen NPS, utilizes two duplicate reactor water clean up systems, each capable of treating 2% of the equivalent reactor feedwater flow. This is significantly better than most BWRs which use typically only 1% systems or the ABWR Certified which uses two 1% system combined into a 2% system.

Finally in the area of effluent releases, the primary gaseous releases are offgas system releases and are evaluated using the models given in NUREG-0016. In this case, for annual releases, the offgas release rate is assumed to be 15% of the design release rate which represents an annualized release rate and is used for comparison to the requirements of 10 CRF 50, Appendix I. This compares to current newer BWRs operating in the same power range and using the same basic systems. For instantaneous releases, comparisons to 10 CFR 20 airborne concentrations are used assuming an offgas release rate four times the design release which represents the maximal value under Standard Review Plan 11.2 (NUREG-0800) at which the reactor can operate for short periods only and beyond which the reactor must be immediately shut down. The offgas system is then sized to meet these offsite dose limitations given these release rates. For water releases, the Lungmen NPS is designed to not release liquid effluent on a regular basis. Since it is recognized that under conditions of high water inventory, some effluents may be released, it is assumed for administrative control that up to 3,700 MBq (0.1 Curie) may be released annually for inventory control.
No changes to the PSAR will be made as a result of the response to this addendum Question.

The following design basis values are based upon GE Policy, safety standards, source term documents, and published SSARs. The expected values are engineering judgment only and are based upon existing state of the art in fuel design and plant operations. As such, the expected values do not represent any warranty or guarantee of GE. No verification or implication of accuracy is implied by publication of these values.

	BWR-4		**BWR-6**		**Lungmen**	
	Design	Expected	Design	Expected	Design	Expected
Noble Gas	100,000 μCi/s	50,000 μCi/s	100,000 μCi/s	25-50,000 μCi/s	100,000 μCi/s	15,000 μCi/s
Iodine	700 μCi/s (I-131)	250-300 μCi/s	700 μCi/s (I-131)	100-150 μCi/s	700 μCi/s (I-131)	100 μCi/s
N-16	50 μCi/g	38-44 μCi/g	50 μCi/g	38-44 μCi/g	50 μCi/g	38-44 μCi/g
Tritium	0.01 μCi/g	0.01 μCi/g	0.01 μCi/g	0.01 μCi/g	0.01 μCi/g	0.01 μCi/g
Ar-41	no specific design basis					
Others	ratio to I-131 from 1971 Source term document		ratio to I-131 from 1971 Source term document		ratio to I-131 from ANS18-1 Source term document	

RESPONSES TO ROC-AEC's PSAR QUESTIONS

ROCAEC Review Comment:

GE is requested to provide a comparison table as follows:

	ABWR		**Lungmen**	
	Design	Expected	Design	Expected
Noble Gas	100,000 μCi/s	15,000 μCi/s	100,000 μCi/s	
Iodine- 131	700 μCi/s	100 μCi/s	700 μCi/s	
N-16	50 μCi/g	38-44 μCi/g	50 μCi/g	
Tritium	0.01 μCi/g	0.01 μCi/g	0.01 μCi/g	
Ar-41	No Specific Design Basis		No Specific Design Basis	
Others	ratio to I-131 from 1971 Source term document		ratio to I-131 from 1971 Source term document	

Further Clarification:

	ABWR		**Lungmen ***	
	Design	Expected	Design	Expected
Noble Gas	100,000 μCi/s	15,000 μCi/s	100,000 μCi/s	15,000 μCi/s †
Iodine- 131	700 μCi/s	100 μCi/s	700 μCi/s	100 μCi/s †
N-16	50 μCi/g	38-44 μCi/g	50 μCi/g	38-44 μCi/g †
Tritium	0.01 μCi/g	0.01 μCi/g	0.01 μCi/g	0.01 μCi/g †
Ar-41	No Specific Design Basis		No Specific Design Basis	
Others	ratio to I-131 from 1971 Source term document		ratio to I-131 from 1971 Source term document	

† Per engineering judgment, the use in the Lungmen NPS of cascaded condenser drains and dual RWCU trains, each with a 2% treatment capacity,may further reduce the expected radioactivity releases. GE will

not be providing expected values for Lungmen that are lower than those expected for the ABWR Certified Design. GE's experience with such predictions is that the methodology involved in generating expected radioactivity concentrations has a large amount of uncertainty in it to make such parameter comparisons come out virtually the same when considering nearly identical designs. The expected values will be revised by the utility after sufficient operating data is available.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: R-11-11

PSAR Sections: 11.5

Question Date: December 16, 1997

PSAR Question:

From Table 11.1-1, Kr-85 is a key nuclide (longest half-life). Please explain why Table 11.5-2 does not include Kr-85 ? Also, Kr-85 is not easy to be absorbed and detected at room temperature. Please explain the methodology for Kr-85 detection.

Response:

The various entries in Table 11.5-2 are intended to list a typical radioisotope for each channel that the radiation detector is capable of sensing and not necessarily all those that the detector is in fact capable of sensing. Certain Regulatory Guides, such as R.G. 1.97, allow the expression of effluent radioactivity in terms of concentrations of Xe-133 equivalents. Additionally, Kr-85 is not listed since many vendors specify the sensitivity of their equipment in terms of Xe-133 for noble gas detection. Based on the above, the detection capability is listed for Xe-133 but not Kr-85. Nevertheless, in all instances where Xe-133 is listed in PSAR Table 11.5-2, an additional entry of "3.7E-3 to 3.7E9 Bq/m^3" will be inserted into Column 3 "Dynamic Detection Range". Column 4 "Principal Radionuclides Measured" will be modified to add "Kr-85" for each aforementioned new entry in Column 3..

The change will be made in the final revision of the PSAR. The capability of Kr-85 detection will be specified in the radiation detector purchase specification.

The exact methodology for the detection of Kr-85 will depend upon the vendor that is ultimately chosen for the supply of the particular radiation channel. Some of the detection principles for Kr-85 would depend upon the existence of the approximately 500 keV gamma ray, while others would search for the 670 keV beta particle. The radiation detector chosen, such as a Geiger Mueller tube or scintillation detector, would depend on what type of radiation the vendor chooses to detect.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **R-11-12**

PSAR Sections: **11.5**

Question Date: **December 16, 1997**

PSAR Question:

In Table 11.5-1, why some units used were mSv/h and others cpm ?

Response:

The selection of the engineering units associated with a particular radiation monitor was based primarily both on previously supplied equipment and the intended use and function of the monitor. As the detailed design progresses, the indicated engineering units may be modified as necessary to ensure optimum selection but in all cases the detection channels will have the required sensitivity and range as stipulated by the appropriate regulatory standards.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: R-11-13

PSAR Sections: 11.5

Question Date: December 16, 1997

PSAR Question:

In Table 11.5-2, the column for Principal Radionuclides Measured, some listed Noble gases, fission products, coolant activation gases. Why not specific nuclides listed ? Also, in Tables 11.5-2 and 11.5-3, which power plants have been considered for the Expected Activity values ? Does it include ABWRs at K-6/7 in Japan ?

Response:

Specific nuclides are not listed for some radiation monitors because the detection channel is typically looking for a large array of radioisotopes and not necessarily for a small specific set. For example, the Main Steam Line Radiation Monitor subsystem is looking for a generalized increase in radiation and is not necessarily concerned with which specific isotopes are contributing to the increase.

The reported activities were derived over a lengthy period of time and are described in Reference 11.1-4 of the PSAR. The domestic nuclear plants involved in this study included the following BWR facilities:

- Pilgrim
- Duane Arnold
- Oyster Creek
- Dresden
- Millstone
- Quad Cities
- Cooper
- Browns Ferry
- Fitzpatrick
- Monticello
- Vermon Yankee
- Nine Mile Point 1

The values do not include radiation release data from K6/7 since the values shown were estimated prior to compilation of long-term operation of the Japanese units.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **R-11-14**

PSAR Sections: **11.5**

Question Date: **December 16, 1997**

PSAR Question:

Regarding the range of channel measurement and display, most systems only consider dose rate mSv/hr, and some systems only consider Cs-137 (e.g., Section 11.5.2.2.5), or Xe-133, Cs-137, I-131, H3, etc. (e.g., Section 11.5.2.2.4). Please explain why some systems (e.g., Section 11.5.2.1.3, Section 11.5.2.2.8 and Section 11.5.2.14) only consider mSv/hr but not I-131 or particulates ? Please provide the rationale for each system.

Response:

Although Standard Review Plan 11.5 does not require that information pertaining to Process Radiation Monitoring (PRM) channel measurement and display be discussed until the submittal of the FSAR, descriptions of these parameters was included to provide basic information on the capability of the to-be-supplied equipment.

PRM Subsystems that provide indication in terms of mSv/hr or Sv/hr are done so typically since the overall radioactivity concentrations of interest are converted to dose rates and are used to provide trip interlock signals. Information about specific radioisotopes is not required for the proper functioning of the channel. These types of monitors include the Main Steam Line Radiation Monitors, the Reactor Building HVAC Radiation Monitors, the Fuel Handling Area Ventilation Exhaust Radiation Monitors, the Control Building HVAC Intake Radiation Monitors and the Drywell Sumps Discharge Radiation Monitors.

Radiation Monitoring Subsystems such as the Offgas Pre-Treatment Radiation Monitor, the Charcoal Vault Radiation Monitor, the Turbine Building Ventilation Exhaust Monitors, the high range Standby Gas Treatment System Radiation Monitors, the Access Control Building Ventilation Radiation Monitor, the Technical Support Center Ventilation

Radiation Monitor and the Containment Overpressure Protection System Radiation Monitors also indicate in either mSv/hr or Sv/hr since the discrete information concerning iodines and particulates is not a regulatory requirement . Thus, in none of the above cases where only mSv/hr or Sv/hr displays are utilized is discrete information pertaining to either iodine or particulate concentrations needed and is therefore not provided.

PRM Subsystems that are used to provide information about effluent releases, such as the Ventilation Stack Discharge Radiation Monitors, the Radwaste Building Ventilation Exhaust Radiation Monitors, the low range of the Standby Gas Treatment Radiation Monitors and the Plant Stack Discharge Radiation Monitors are provided with additional means to ascertain both iodines and particulates. This information is needed to demonstrate compliance with the applicable regulatory requirements for effluent releases.

The Offgas Post-Treatment Radiation Monitoring Subsystem provides indication of iodines and particulates in order to access the functionality of the Offgas Charcoal Treatment process. The Reactor Building Cooling Water Radiation Monitors and the Radwaste Liquid Discharge Radiation Monitor are specified in terms of a reference isotope, Cs-137, since this is a typical radioisotope for defining sensitivities of liquid monitors. The Drywell Fission Product Radiation Monitoring Subsystem measures particulates in order to comply with the intent of Regulatory Guide 1.45, as it concerns Reactor Coolant Boundary leak detection.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: R-11-15

PSAR Sections: 11.5

Question Date: December 16, 1997

PSAR Question:

In page 11.5-11, 2nd paragraph, 2nd line which reads "The high-high upscale trip and the downscale/inoperative trip are used to stop the discharge to the environment" and Section 11.5.2.1.1 (in page 11.5-5), 4th paragraph, 3rd sentence which reads "any two-of-four channel trip results in ...." are not very clear as to what they mean. Please explain the actions required after each of the conditions : high-high, high, downscale, inoperative trip.

Response:

The various trip setpoints are correlated with different radiation levels of interest. These levels will be determined during the preparation of the Offsite Dose Calculation Manual (ODCM) and will have unique trip points associated with them.

In the case of the single channel Radwaste Liquid Discharge Radiation Monitor , i.e., subsection 11.5.2.2.5, the high-high trip is used to initiate closure of the isolation valve on the discharge and will be correlated to an acceptable radioisotopic release concentration. The high trip serves as a warning to operator that the process may approaching a radiation level of significance and that some manual action might be appropriate. The downscale and/or inoperative trips are used by the internal circuitry of the radiation monitor to indicate potential problems with its electronic functions. These trips, depending on logic configuration chosen, can also be used to precipitate valve isolation and alarm if desired.

In reference to the "any two out of four channel trip" for the four channel Main Steam Line Monitor described in subsection 11.5.2.1.1, the logic will provide an initiation signal on receipt of two high-high trips received from any of the four channels. The downscale and/or inoperative trips can be used by the internal circuitry of the radiation monitor to indicate potential problems with its electronic functions. These trips, depending on the logic configuration that is selected, can also be used with the high-high trips, if desired, to precipitate the Mechanical Vacuum Pump actions described in the text.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: R-11-16

PSAR Sections: 11.5

Question Date: December 16, 1997

PSAR Question:

In page 11.5-15, Section 11.5.3.1 stated that "10CFR20 Limits" should be complied. Why not the ROC Standards of Radioactive Protection Safety been used instead ? Questions like this should consider the local regulations first.

Response:

Subsection 11.5.3.1, last sentence will change to read "Monitoring of each major path provides radiation measurements that enable the demonstrate of compliance with the ROC document entitled "Standards of Radioactive Protection Safety" dose limits for the General Public due to effluent releases. Table 1.8-21 will be modified to include ROC document "Standards of Radioactive Protection Safety" dated July 10, 1991."

The change will be made to the final revision of the PSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: R-11-17

PSAR Sections: 11.5

Question Date: December 16, 1997

PSAR Question:

In page 11.5-15, Section 11.5.3, items (1) & (2), it was stated that "Liquid and Gaseous releases are monitored for gross gamma radioactivity". Should only gross gamma be monitored ?

Response:

Subsection 11.5.3, items (1) and (2) should each read "Liquid releases are monitored for radioactivity as listed in Table 11.5-3." and "Gaseous releases are monitored for radioactivity as listed in Table 11.5-2.". The final revision of the PSAR will be changed to incorporate the above description.

Radiation measurement is not limited to gross gamma.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **R-11-18**

PSAR Sections: **11.5**

Question Date: **December 16, 1997**

PSAR Question:

In page 11.5-19, Section 11.5.5.4, please explain why "Audits and verification during normal plant operation are out-of-scope for the Lungmen NPS" ?

Response:

The sentence should be modified to state that "Audits and verification during normal plant operation will be conducted in accordance with Sections 13.4 and 13.5." This would be in keeping with the intent of the Standard Review Plan for Chapter 11.5, Review Area I.2.

The above defined change will be made to the final version of the PSAR as a result of this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: R-11-19

PSAR Sections: 11.5

Question Date: December 16, 1997

PSAR Question:

In Table 11.5-6, nuclides such as Sr-89, Sr-90 were sampled from the liquid discharge every 3 months. Is it possible the sampling results show high activity of Sr-89/Sr-90 but the discharge is already released outside the plant ? If yes, how can it be prevented from happening ?

Response:

Per Subsection 11.2.1.2, releases via the liquid radwaste discharge line will be done via a batch release. The activity in this line is monitored by a radiation monitor whenever a release is in progress. In addition, Subsection 11.2.3.2 provides further information on the means to control the release of liquid effluents to the appropriate regulatory levels. It is therefore not considered probable that the liquid sampling results would show high activity of Sr-89/Sr-90 without having a prior isolation thus precluded a release.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： R-11-020

問題章節(PSAR Section)： Section 11.6

初提日期(Question Date)： 1997.12.16

問題內容(PSAR Question)：

11.6 乙節表示 Offsite Radiological Monitoring Program 將於 FSAR 提供，對此有下列意見：

(1)依本會「環境輻射偵測規範」，核設施在運轉前之調查須實施二年以上，俾使核設施於運轉後執行環測時可熟練運作，為此若於 FSAR 提出是否時程上將不相符。

(2)運轉前的背景調查在於取得有關之關鍵性核種、關鍵性曝露途徑及關鍵群體等資料，作為運轉後環測計畫的擬定及評估劑量的參考資料，而有關此類調查結果見諸於何章節，或其調查將於何時進行。

(3)為了解核設施運轉後之影響，更需有運轉前環境輻射及環境試樣的背景強度及特性。

而依「核能四廠第一、二號機發電計畫環境影響評估報告」之審查結果，亦曾要求台電公司對「環境劑量評估所需本土基礎資料，包括電廠附近居民生活及飲食習慣、人口分佈狀況及土地利用等應確實調查建立」。且亦要求台電公司應「研擬更具體與完整之環境監測計畫，務期能建立完整之環境背景資料庫，確實掌握電廠施工與運轉對環境的實際影響」。

問題答覆(Responses)：

一、有關核四廠環境影響輻射偵測，將遵照原能會頒佈之「環境輻射偵測規範」，於燃料裝填前兩年執行偵測作業。

二、有關關鍵核種之資料已列於 PSAR 第十二章 Table 12.2-19 及 Table 12.2-22；另外，有關民眾劑量評估及曝露途徑，將比照現行核一、二、三廠參照美國 R.G. 1.109 "Calculation of Annual Dose to Man from Routine Releases of Reactor Effluent for the Purpose of Evaluating Compliance with 10CFR Part 50, Appendix I"及原能會頒佈之「核設施環測結果民眾劑量估算導則」執行。

三、有關核四廠背景環境輻射調查，將於燃料裝填前兩年執行背景輻射

偵測作業，以瞭解當地運轉前輻射背景及環境試樣特性。

四、為配合核四廠環境影響評估(EIA)，鹽寮地區環境輻射偵測作業自民國七十年二月起執行至八十年九月，長達十一年，其環境輻射資料相當完整，並收錄在環境影響評估報告第二章第九節。

五、有關核四廠運轉前附近居民生活及飲食習慣、人口分佈及土地利用資料調查，將配合燃料裝填前兩年執行之背景環境輻射偵測作業，委請學術單位調查。

台電公司進一步澄清說明：

有關核四廠環境輻射偵測，將遵照原能會頒布之「環境輻射偵測規範」於運轉前三年提報「環境輻射偵測計畫」送原能會審查，並依規定於運轉前二年執行偵測，詳細時程表如下頁：

核四廠環境偵測計畫執行時程表

分項工作	預定執行進度										備註
	88/1	88/7	89/1	89/7	90/1	90/7	91/1	91/7	92/1	92/7	
核四廠歷年實數據整理											核四環境評估報告及相關資料整理
核四廠設計相關資料及環境評估資料收集											核四環境評估報告及相關資料整理
核四相關環境規劃研究計畫發包											
執行核四相關環境規劃及修訂											89年計畫於8月送審
環測對照站選擇及環境試樣取樣實測分析											
鹽寮地區環境試樣取樣實測分析											依法規需於運轉前兩年執行偵測

註： 以上之時程規劃係以核四廠於92年8月進行燃料裝填為前提而訂定，如實際狀況變更，將配合修正。

PSAR第11.6節Offsite Radiological Monitoring Program之內容將改寫如下：

Lungmen NPS Offsite Radiological Monitoring Program will follow ROCAEC's "Technical Specification for Radiological environmental Monitoring". The program will be submitted to ROCAEC for review and approval three years before fuel loading and will be implemented two years before fuel loading. The detail planned schedule and activities are shown in the following figure:

Planned schedule of
Lungmen NPS Offsite Radiological Monitoring Program

Activities	Planned Schedule										Remarks
	1999 Jan.	1999 July	2000 Jan.	2000 July	2001 Jan.	2001 July	2002 Jan.	2002 July	2003 Jan.	2003 July	
Reveiw Historical Data											
Background Data Collection											
Study of Offsite Radiological Monitoring Program											
Conduct and Modify of Offsite Radiological Monitoring Program											The Program will be submitted to ROCAEC for review and approval on August 2000
Implementation of Offsite Radiological Monitoring Program											

Note: Planned schedule is based on the fuel loading date of August 2003.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: F-11-001

PSAR Sections: Chapter 11

Question Date: December 6, 1997

PSAR Question:

1. Seismic designs are mentioned in Sections 11.2.1.2.2 and 11.4.1.4. Please clarify if the effect of horizontal and vertical acceleration need to be considered concurrently for seismic design of major structures of equipment (Section 11.3.7 of PSAR states that the seismic design of charcoal absorbers and their support elements are based on 0.2g horizontal and vertical accelerations).

2. For radioactive gas waste treatment system, please clarify if there is a discharge piping designed for a proper retention time. For the design of charcoal absorbers described in PSAR Section 11.3.3.3.12, please provide information regarding adequacy of absorbing retention time for effectively reduce the radioactive release of inert gas. (similar system of TPC Second Nuclear Power Plant possessed ten (10) minutes retention discharge piping design; the charcoal absorber has capabilities of retaining Kr for 46 hours and Xe for 42 days, so that the inert gas sufficiently decayed.).

3. Fig. 11.3-1 does not show upstream prefilter while similar system of TPC 2nd Nuclear Power Plant has. Please clarify; if not, please explain why it is not required.

PSAR Response:

1. The buildings housing the Radwaste systems will at a minimum meet the seismic requirements of Regulatory Guide 1.143. For detailed design requirements, please refer to Chapter 3 Section 3.8.4.5.3. Regulatory Guide 1.143, Regulatory Position 5, provides guidance for the determination of seismic loads for seismic design for radwaste management systems and structures housing radwaste management systems. Regulatory Position 5

recommends the use of Regulatory Guide 1.60 to define the seismic ground motion for design purposes which is defined in terms of two horizontal and one vertical component. This implies that the effects of horizontal and vertical accelerations should be considered concurrently for seismic design purposes.

In accordance with Regulatory Guide 1.143 ,Regulatory Position 1.1.4, equipment and components used to collect, process, and store liquid radioactive waste need not be designed to the seismic criteria given in Regulatory Position 5 and therefore, the effects of horizontal and vertical accelerations need not be considered concurrently. Similarly, in accordance with Regulatory Position 3.1.4, equipment and components used to collect, process, and store solid radwastes need not be designed to the seismic criteria given in regulatory position 5 and therefore, the effects of horizontal and vertical accelerations need not be considered concurrently.

However, in accordance with Regulatory Position 2.1.3, those portions of the gaseous radwaste treatment system that are intended to store or delay the release of gaseous radioactive waste should be designed to the seismic design criteria given in Regulatory Position 5 and therefore, the effect of horizontal and vertical accelerations should to be considered concurrently. The seismic design in Sections 11.2.1.2.2 and 11.4.1.4 will be revised as noted in Attachment A.

(The seismic design in Sections 11.2.1.2.2 and 11.4.1.4) will be changed to the PSAR page 11.2-3 and 11.4-4.

2. A delay pipe is not required in the Lungmen design as the system is not operated on a bypass. Sections 11.3.1.2 states that charcoal absorbers have the capability of retaining Xe for 60 days.

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

3. No additional filtration is required as the system is not designed to be operated in a by-pass mode. The charcoal beds will act as a filtering medium.

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

ATTACHMEMT A

11.2.1.2.2 Seismic Design

The buildings housing the liquid radwaste process equipment are designed as Seismic Category IIB. The base mat and outside walls are designed seismically to the Operating Basis Earthquake in accordance with regulatory position 1.1.3 of regulator Guide 1.143 or ROC Building Code "Medium seismicity zone", whichever is larger, to a height necessary to retain spilled liquids within the building.

Dikes and retention basins for outdoor liquid radwaste tanks shall be capable of preventing runoff in the event of a tank overflow and should have provisions for sampling collected liquids and routing them to the LRWPS. These dikes or retention basins are designed to the same retention requirements in Regulatory Guide 1.143 as the Radwaste Facility.

11.4.1.4 Seismic Design

The SRWPS equipment is not required to be designed to withstand the effects of a seismic event. The foundations and adjacent walls of the Radwaste Facility housing the SRWPS equipment shall be designed seismically to the Operating Basis Earthquake in accordance with regulatory position 3.1.3 of Regulatory Guide 1.143 or ROC Building Code "Medium seismicity zone", whichever is larger, to a height sufficient to contain the maximum liquid inventory expected to be in the building.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: F-11-002

PSAR Sections: Chapter 11

Question Date: December 6, 1997

PSAR Question:

1. In general, during the reactor start-up, there is no sufficient steam to drive the SJAJ. Hence, the mechanical vacuum pumps are needed to be operated. PSAR Fig.11.3-2 shows no mechanical vacuum pumps are operated during plant start-up. Please explain the system design and its operation.

2. TPC expects to have hollow fiber filters as pre-filters in the condensate polishing system. It seems pre-coat or other type of pre-filters are recommended and spent resins are not regenerated. Would this increase the rate of producing suspended solids/resin?

PSAR Response:

1. The mechanical vacuum pumps are included in the Main Condenser System. They will be used to draw down the condenser prior to transferring to the steam jet air ejectors. The mechanical vacuum pumps will discharge directly to the plant stack located immediately adjacent to the respective unit switchgear building.

2. In accordance with prevailing industry design practice, Lungmen is committed to utilizing backwashable filters with no precoat in the condensate demineralizer system and the condensate demineralizer system will not regenerate resins . The partially expended resins from the demineralizer beds will be transferred to the RSWS for further use in the LRWPS prior to volume reduction and solidification.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **F-11-003**

PSAR Sections: **11.1, 11.2, 11.3**

Question Date: **December 10, 1997**

PSAR Question:

1. Sections 11.1 and 11.1.1.2 state the core release rate of I-131 are 24.4 and 25.9 MBq/s respectively, are inconsistent. Which one of these is Lungmen design basis value?

2. 3700 MBq/s liquid per Unit will be released annually. Will the radioactive waste be accumulated at a release point near the plant after 40 years of plant operation?

3. Method of using which equipment to treat off gas be released from radwaste building stack and reactor building stack via ventilation release stack need to be described.

4. How often the 200 tons of charcoal need to be refilled?

5. The meaning of " ventilation release 50 C " in Section 11.3.9.2 needs to be explained.

6. The parameters " karb for Kr and Xe at 25°C at least 39 and 1160 cm/g respectively " shown in Section 11.3.1.2 are not consistent with those in Section 11.3.3.2.4.

7. Please explain why Section 7.11.3.2.1 (5) is 38°C.

8. Please explain why the value Xe-131m5.5E-01 MBq/s shown in Table 11.3-3 is different from that in Table 11.1-1.

PSAR Response:

1. This is a typographical error. From the context of the section the correct number is 25.9 MBq/s.

2. Effluent from the Liquid Radioactive Waste Processing System (LRWPS) will be released through the Circulating Water discharge tunnel into a multiport diffuser. The discharge diffuser jet exiting with a high exit velocity with mix rapidly with the ambient ocean water to achieve a large dilution in the near-field. The discharge plume will be further dispersed by the ocean current in the far-field. Due to the dynamic interaction of the plume with local conditions and the refresh rate established by contiguous currents and ocean storms any deposition in the bay sediment will be minor with no anticipated impact on the Appendix I analysis.
3. Responses as below:

3.1 The Radwaste Building will be equipped with a process vent stack which will handle the following system flow streams:

- Radwaste Building Ventilation Exhaust
- Radwaste Tunnel Ventilation Exhaust
- Radwaste(Solid & Liquid) System Process Vents
- Incinerator Exhaust

All flow streams will be processed through HEPA filters.

3.2 The Plant Stack ,located immediately adjacent to the Switchgear Building, takes effluent flow streams from the following sources; 1)Turbine Building Ventilation Exhaust; 2) Reactor Building Secondary Containment HVAC Exhaust; 3)Gaseous Radwaste Waste Process System; 4)Mechanical Vacuum Pumps. The Turbine Building Exhaust is processed through HEPA filters while the GRWPS is equipped with HEPA filters downstream of the charcoal beds. The Mechanical Vacuum Pumps currently exhaust directly to the Plant Stack with no filtration or mechanical delay. These pumps are not currently designed to operate while nuclear steam is admitted to the condenser. The Reactor Building Secondary Containment Exhaust system exhausts to the Plant Stack through 80-85% filters(ASHREA52). The Reactor Building Stack services the Standby Gas Treatment System which processes its exhaust through HEPA filters and a charcoal delay bed.

4. The design intent of the GRWPS is to never replace the initial charcoal load . The system layout is designed to accommodate a change out, but none is anticipated. If water intrusion into the charcoal beds occurs, the system is designed for the "in situ" drying of the wetted charcoal.

5. The temperature of the combined flow rates up the plant stack referenced in section 11.3.9.2 will not exceed 50°C.

6. The values should both be defined as 60 and 1170 respectively.

7. The reference to 38°F should be deleted . The environmental parameters associated with the GRWPS are more appropriately described in section 11.3.3.3.3.

8. This is a typographical error. The value in question should be changed to
7.4E-01.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **F-11-004**

PSAR Sections: **11.4, 11.5**

Question Date: **November 30, 1997**

PSAR Question:

1. In order to determine the reactivity of the drum in which the solid waste is to be contained prior to solidifying the waste, isn't it necessary to take sample for isotopic (including α-decay) and reactivity analysis for wet waste and dry active waste prior to solidification?
2. The inspection of the all the radwaste storage drums (not only the compressed drums by the supercompactor), needs the following requirements:
   2.1 Drum surface smear test (to use remote control smearing).
   2.2 Detect quantity of surface dose rate.
   2.3 Detect the density and type of the radioactivity.
   For the solidified waste storage equipment, should evaluate the usage of the automatic detect equipment for the waste drum.
3. In page 11.4-5, described that solidification agent storage less than 25 gy/hr as a low radiation area, but the 25 gy/hr should not be a low radiation area.
4. In page 11.4-20 Figure 11.4-1, the secondary path is incomplete, also does not specify when it will be used.

PSAR Response:

1. See Attachment F-11-004-1.
   No changes will be made to the PSAR as a result of the response to the question.
2. See Attachment F-11-004-1.
   No changes will be made to the PSAR as a result of the response to the question.
3. The PSAR section contains a typographical error and should be changed from 25Gy/h to 25μGy/hr.
4. Fig 11.4-1 has been updated as attached to address this issue. The design provisions of secondary path are for operation flexibility to provide additional treatment route or backup of normal treatment

process. The actual process of secondary path will be finalized during the design stage subsequent to Radwaste system procurement.

Fig 11.4-1 will be added to this PSAR section.

Attachment F-11-004-1

Dry Active Waste

The Dry Active Waste processed through the compaction system will have it's radioactivity monitored, tracked and inventoried utilizing a system and components to those described below.

Drum Surface Radiation Monitoring System

A stationary radiation monitoring system (range: 1 r/hr - 10^4 r/hr, accuracy: ± 15% of full scale when calibrated with Co-60) shall be provided and installed outside of the enclosure to permit determination of the radiation levels at the surface of the filled overpack drum and at a predetermined point within 3 feet of the surface of the overpack drum.

Radiation monitoring equipment shall also be provided to permit making an isotopic analysis (both quantity and type of radioactive material present) of each waste container and to permit its classification in accordance with ROC-AEC rules.

The collected data (surface dose rate and isotopic analyses) plus the results of the surface contamination measurement shall be automatically transmitted to the computer system for processing.

The monitor must also interface with the monitoring system provided by the NI Supplier..

Means shall also be provided to permit obtaining samples periodically for determination and verification of the scaling factors employed for waste classification. The sampling provisions shall minimize radiation exposure to operating personnel.

Smear Test and Decontamination System

An automatic mechanism for remote retrieval of a wipe sample at an spot on the exposed surface of a filled overpack drum shall be done remotely and automatically or semi-automatically in order to minimize radiation exposure to the operator. The wipe test sample will be monitored in an area aside from the supercompactor room.

A decontamination station with a mechanism which will facilitate dry-wipe decontamination of the filled overpack drum shall be provided.

A handglove box equipped with a turntable and elevator to achieve a remote smear test and *manual decontamination of any point of the overpack drum lateral surface shall be provided.*

A separate control panel shall be provided and mounted locally to perform all the operations inside the handglove box.

Drum Height and Weight Measuring System

The system shall be capable of measuring the height of the crushed input drum and the weight of the input drum before crushing and transmitting the measured data to the computer system for processing. Height sensing shall be accomplished by employing a non-contacting transducer. The fill height and weight of overpacks shall be measured as well as the weight of solidified/conditioned drums.

Optical Surveillance Facilities

The Supplier shall design and arrange the operation to permit direct visual surveillance of all functions performed from the control room. If direct full visual surveillance of the operation of the supercompaction system from the control room is not possible, a CCTV (closed circuit color television) monitoring system shall be provided to complement visual observation.

The windows of the control room shall be designed according to the dose rate anticipated from DAW, the windows in the control room should provide a field of clear vision of major equipment.

Computer System

The computer system will receive the following information:

1. *Message/commands from the computer keyboards*
2. *Data from the Drum Height & Weight Measuring System*
3. *Data from the Drum Surface Radiation Monitoring System*
4. *Data from the Drum Isotopic Analysis Monitoring System*
5. *Data from the Offgas Radiation Monitoring and Recording System.*

Data received from the Drum Height & Weight Measuring system shall be processed in the computer so that

1. *The crushed input drum will be distributed, through the selection of the computer, and loadedto the selected overpack drum so that the overpack drum volumetric capacity may be maximally utilized.*
2. *The Fill height and total drum weight for the loaded overpack drum shall be recorded in the computer.*

In addition to receiving and recording the information identified above and optimizing the loading and fill height of the overpacks, the computer system shall be capable of generating reports in hard copy in English (Chinese Character Display Capability). The report shall at least include the following information:

Input Drum

- *Date drum received*
- *Origin (from which nuclear power unit)*
- *Drum Serial number*
- *Drum weight*
- *Surface radiation dose rate*

Overpack Drum

- *Date drum processed*
- *Origin (from which nuclear power unit)*
- *Drum Serial number*
- *Drum weight*
- *Surface radiation dose rate*

Radioactivity levels of the off-gas

Solid Waste Processing System

The drums processed through the SWRPS will be monitored, tracked, and inventoried by a system and components similar to those described below

Container Surveying

Means and equipment shall be provided to permit making a surface dose radiation survey of each solidified or filled drum and measurement of the concentration, quantity and type of radioactive material present in each waste container to permit its classification in accordance with the rules of the ROC-AEC. This survey and quantitative measurement of radioactivity shall be accomplished without requiring removal of the waste contents from the container or from waste feed lines to the container. The collected data shall be *automatically transmitted to the computerized inventory control system part of the SRWPS.*

Means and equipment shall be provided to permit obtaining samples periodically for the determination and verification of the scaling factors employed for waste classification. Such sampling provisions shall consist of either a sampler for extracting a sample from slurry transfer lines or means for removing a sample from storage tanks using a long-handled tube or scoop. The sampling provisions shall minimize radiation exposure to operating personnel.

Means and equipment (such as smear testing and decontamination equipment, etc.) shall be provided in order to control absence of surface contamination of the drums and to decontaminate them if necessary. Results of the surface contamination measurement shall be transmitted to the computerized inventory control system. The means shall be designs to be operated remotely and automatically or semi-automatically to minimize radiation exposure to the operator.

Computerized Inventory Control System (CICS)

The CICS shall include the following software, provided in a form that can be read directly by the provided hardware. Source code for any proprietary programs shall be provided to TPC at completion of field testing and system turnover.

- *Computer operating system, complete with any data transfer protocol required*
- *Applications programs for data entry, in English or Chinese, computation of drums current isotopic radioactivity and surface dose rates, data retrieval, data archival and report generation in English or Chinese*
- *Used program languages*
- *System used and reference manuals, including complete instructions for any special hardware/software necessary for Chinese language input/output*
- *Inventory control program user's guide, including complete instructions for data manipulation in Chinese or English*
- *Programmer's guide for the system and provide program language*
- *Printer user's guide*
- *source code of the programs.*

The CICS shall assure that the following information regarding each and every drum within the facility shall be available for entry, retrieval, and display at the terminal, printing on the printer or archival on tape or disk:

- *Drum identification*
- *Drum Storage location*
- *Drum surface dose rate (initial)*
- *Drum surface dose rate (current)*
- *Drum isotopic content for initial and current (20 isotopes minimum)*
- *Date stored*
- *Date of expected removal or date of removal*
- *Drum gross weight*
- *Date of solidification*
- *Radwaste source, composition (includes solidification agent, radwaste, etc)*
- *Remarks.*

The CICS shall include the following features as a part of its standard operating mode:

- *Automatic Data archival upon system shutdown*
- *Automatic data archival and system archival upon loss of power*
- *Capability for generating reports in English or Chinese on sorted drums within the facility for the information parameters identified above*
- *Capability for interpreting, compiling and running a program writer in the provided language*
- *All operations initiated by operator in response to prompts on the CRT screen (operator/computer dialog*

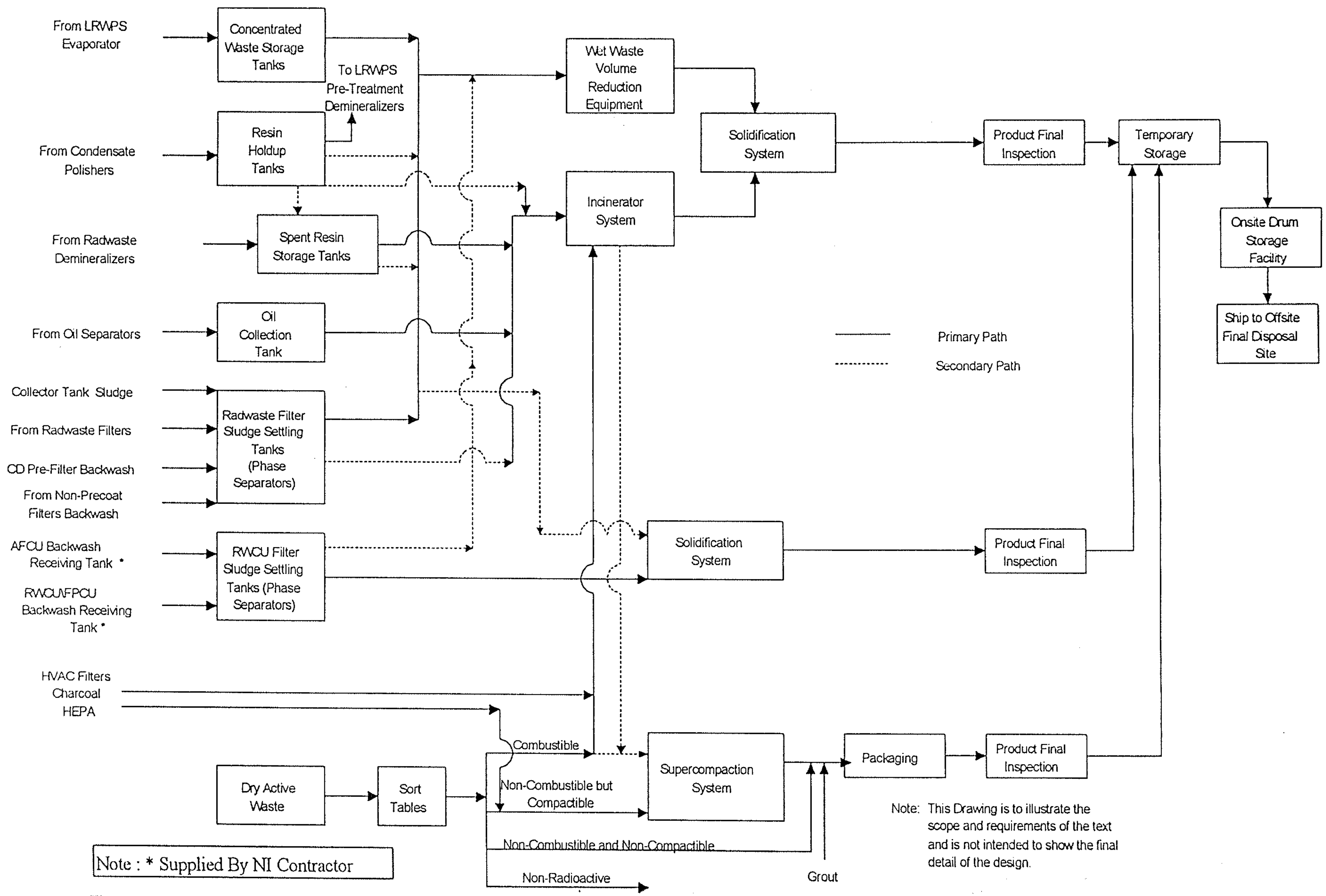

Figure 11.4 - 1 SRWPS Process

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: F-11-005

PSAR Sections: Chapter 11

Question Date: December 9, 1997

PSAR Question:

1. Regarding the computer model which is used to evaluate the diluted condition of both Gaseous Radioactive Waste Processing System (GRWPS) and Liquid Radioactive Waste Processing System (LRWPS), how to validate this model is practical and accuracy in Taiwan area. Please provide the further explanation. (Ch. 11.3)

2. Is there any other type of vessel can be used for Radwaste storage system, except the 55 gallons drum? Please provide the further explanation based on the associated code. (Ch. 11.4)

3. Please provide the clear explanation for the design concept of processing for spent resin (such as Volume reduction or solidification). (Ch. 11.4)

4. Does the solidification system be capable of controlled by the effective QA/QC? How the solidified material be handled if it does not meet the solidification standard? Please explain it. (Ch. 11.4)

PSAR Response:

1. Responses as below:

    1.1 As noted in sections 11.3.9.3&4 the specific analysis for the RGWS will be provided in the FSAR. Preliminary analysis of the Lungmen site meteorology indicates that the meteorological dispersion characteristics for the site are bounded by the ABWR certification.

    1.2 As noted in section 11.2.3,the LRWS effluent releases dilution calculations and validations will be provided in the FSAR.

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

2. The components of the SRWPS are specified as utilizing 55gallon drums. Certain non-compressible wastes will be packaged in special containers. Other container designs are available. The choice of container design is typically predicated upon the utility's operating philosophy and disposal site requirements. In the case of Lungmen the 55 gallon drum is consistent with prevailing operating philosophy and the intended design of the on-site storage facility.

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

3. High activity resins associated with the Nuclear Island clean up systems will be solidified. Low activity resins associated with the Condensate Polishing System and the Liquid Radwaste System will be processed through the volume reduction system incineration and solidification. The partially expended resins of the Condensate Polishing system will be further expended in the Liquid Radwaste System prior to disposal through the SRWPS.

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

4. The Supplier will provide a Process Control Program (PCP)for SRWPS in order to guarantee that those systems will produce conditioned waste that meet ROC-AEC requirements. The format and content of the PCP will meet the requirements set forth by ROC-AEC. The SRWPS system will be capable of solidifying waste to the following criteria:

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: F-11-006

PSAR Sections: Chapter 11

Question Date: November 28, 1997

PSAR Question:

1. In Figure 11.4-1 (SRWPS Process Flow), It did not specify the liquid process design and indicated if there is any liquid existed in the process? For example, if any water is required for the incinerator (quench). Please specify the flow quantity, chemical contents of the liquid, where and how this liquid will be delivered and handled.
2. In the process of the wet waste volume reduction, incineration, RWCU filter sludge settling and super compaction, it will produce the solidified material (drum), please provide the solidified material quantity for every portion of the process during the normal operation.
3. In Table 11.2-2 item LPW described that the hours to process Max. daily rate = 24.7. Please clarify that if the 24.7h is a typo?
4. Portion of the handling process on figure11.2-2 and 11.2-3 is not settled, please submit more detailed data for the design of handling process and estimate of the product quality.

PSAR Response:

1. Water associated with resin transfer will be decanted to the LRWPS HIGH Purity Waste Subsystem as depicted in figure 11.2.1 Any liquid associated with the incineration process, whether quench or scrubber waste, will be forwarded to the LRWPS Low Purity Waste Subsystem as depicted in Figure 11.2.2. The actual process quantities and flow paths will be supplied with the FSAR subsequent to Radwaste System procurement.
2. Process flow Diagrams including the quantity of drum for each kind of wastes, will be provided with the FSAR subsequent to Radwaste System procurement.

3. The table is correct as noted. The evaporators were sized at a nominal 25gpm. When the maximum daily flow is considered the processing time equates to marginally over one day. This mean that the fore tankage inventory will slightly increase. It should be noted that the evaporators are capable of processing a normal daily liquid waste generation in one 8 hour shift.

4. The exact flow paths have been defined subsequent to the initial submission of the PSAR. All waste steams noted on these figures are now routed through the liquid radwaste evaporators.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: F-11-007

PSAR Sections: Chapter 11

Question Date: December 11, 1997

PSAR Question:

1. Please specify the feasibility of recovery and re-use of the treated detergent drain waste water. (Because the yearly radioactivity releasing rate from detergent drain waste water discharge at existing Nuclear Power Plants in Taiwan is increasing)

2. How to determine the activity fraction of every buildings in Table 11.2-3?

PSAR Response:

1. The operating philosophy of recycling the treated laundry waste has not been selected for Lungmen in order to preclude the possibility of contaminating the condensate supply with organics and/or surficants found in detergents. One case of cross contamination would generate large quantities of liquid waste inventory.

2. The activity fractions are based on GE operating experience. The final values and quantities will be supplied with the FSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **F-11-008**

PSAR Sections: **11.4**

Question Date: **December 12, 1997**

PSAR Question:

Please consider the design to use more efficient process system for the solidification and volume reduction.

PSAR Response:

The solidification and volume reduction equipment is being provided in accordance with the design objectives stated in Section 11.4.1.1 of the PSAR. The overall Unit objective of 250drums/yr/unit is a very aggressive design goal when measured in the context of US and European operating experience.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 11-029

PSAR Sections: Ch. 11 Sec. 5

Question Date: May 12, 1998

PSAR Question:

Section 5 of this chapter described "Process and Effluent Radiological Monitoring and Sampling System" but no consideration was given to the PASS. Please clarify.

Response:

There was no consideration given to the PAS (Post Accident Sampling System) in Section 11.5,"Process and Effluent Radiological Monitoring and Sampling System", since the PAS is part of the Containment Monitoring System (T62). The system description for the PAS is described in Section 7.5.1.3, "Containment Monitoring System - Instrumentation and Controls".

No changes will be made to Chapter 11.5 of the PSAR as a result of this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 11-030

PSAR Sections: Ch. 11

Question Date: May 12, 1998

PSAR Question:

In general ABWR will take the following steps for radwaste volume reduction :

(1) Hollow fiber filters used for condensate filtrations.
(2) Non-regenerative particulate resins used for condensate demineralizers.
(3) Incineration for combustible solids and spent resins.

In this chapter, no information was given to items (1) and (3) above. Please provide this information. Also, is the spent resins stated in item (2) included in the incineration as well ? Please clarify.

RESPONSE:

(1) Hollow fiber filters will be the preferred filtering media in the liquid

radioactive waste system and are being considered for use in the condensate polishing system.

(2) Non-regenerative particulate resins used for condensate demineralizers will be partially depleted in this capacity and then transferred to Solid Radwaste system, stored in resin holdup tanks, and re-used in the LRWPS pre-treatment demineralizer for the high purity waste trains. Spent resins from the LRWPS demineralizers will be processed through the incineration system.

(3) The solid radioactive waste processing system (SRWPS) includes wet waste volume reduction equipment, an incinerator system, a waste solidification system and a supercompaction system. All combustible solid waste is expected to be incinerated including charcoal HVAC filters, spent resin (excluding reactor water cleanup resins), waste oil from oil separators and combustible waste extracted at the sorting tables from dry active waste.

No changes will be made to the PSAR as a result of the response to the question

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **12-001**

PSAR Sections: **Ch. 12**

Question Date: **February 16, 1998**

PSAR Question:

What does "Class" mean in Table 12.2-7 and other similar Tables? Please explain.

PSAR Response:

The word "Class" is used to denote a grouping of elements (or in this case isotopes) by their chemical properties. The class grouping method was adopted from ANS Standard 18.1-1984. The classes are defined as:

Class 1	Noble Gases
Class 2	Halogens
Class 3	Cesium, Rubidium
Class 4	Water Activation Products
Class 5	Tritium
Class 6	Other Nuclides

The table above will be added as a note to Table 12.2-7of the PSAR

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **12-002**

PSAR Sections: **Ch. 12**

Question Date: **February 16, 1998**

PSAR Question:

In Table 12.2-18a, why the value for Rotating-ball Spindle is zero Before Cleaning ? and why the values for Throttle Bushing Before and After Cleaning are the same?

PSAR Response:

The values given in Table 12.2-18a were taken from measurements made on an FMCRD at the end of an operational cycle at an operating BWR where a test FMCRD was installed. The values represent only the order of magnitude for the contamination levels which could be expected to be seen. There is no analytical method of calculating contamination levels. GE placed a test FMCRD into the reactor for a single cycle and then removed the drive and measured the radiation rates which are given in column 2 (Before Cleaning). GE cleaned and prepared a treatment bath as would be expected as part of the decontamination effort for FMCRD. However, the bath was inadvertently contaminated before use which resulted in GE using a contaminated bath. Therefore, though some of the values were reduced, other components like the rotating ball spindle were contaminated which is seen by the upward change in radiation rate. Rather then show no "after cleaning" values for the components, the measured values are shown since the actual values after decontamination would be lower with proper decontamination and since the overall radiation contamination at Lungmen is expected to be lower than the test plant due to better materials used. Therefore, it is deemed that the as measured values are conservative. To date, data from the operating ABWRs show little to no contamination in the FMCRDs with only the drive components which are inserted into the core showing high radiation rates.

Table 12.2-18a will be revised in column 2 to show that "before Cleaning" value for the "Rotating-ball Spindle" as "<LLD". Additionally, an

asterisk footnote (*) will be added after the entry for "Rotating-ball Spindle" and "Throttle Bushing". A footnote at the bottom of the table will state: "* The gamma dose values shown here were taken from measurements made during a FMCRD test program. The After Cleaning dose rate (in column 3) of the Rotating-ball Spindle and the Throttle Bushing resulted from a contaminated treatment bath."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **12-003**

PSAR Sections: **Ch. 12**

Question Date: **February 16, 1998**

PSAR Question:

Please explain what "Maximum Technical Specification" represents in Table 12.2-20 and similar Tables.

PSAR Response:

The term "Maximum Technical Specification", as used in PSAR Table 12.2-20, refers to the acceptable radionuclide concentration values found in Table 2 of Appendix B of the revision of the U.S. 10 CFR 20 under which the ABWR was certified in the U.S. In accordance with 10CFR 20.106, external gaseous releases must result in concentrations less than those shown in the aforementioned 10CFR20 table. Therefore plant specifications are required to maintain release rates less than that value as determined on an instantaneous basis (not an annual basis).

This is the only reference to maximum technical specification in PSAR Chapter 12.2.
[NOTE: THE RELEVANT INFORMATION PREVIOUSLY CONTAINED IN THE OLD VERSION OF 10CFR20.106 CAN NOW BE LOCATED IN 10CFR20.1302]
A note will be added to Table 12.2-20 stating that, for column 3, "The term "Maximum Technical Specification" refers to the acceptable radionuclide concentration values found in column (7) and (8) of Table 4 of the revision of the Republic of China regulation entitled "Standards for Protection Against Radiation" and Table 2 of Appendix B of U.S. 10CFR20."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-004

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

In Table 12.2-21 Part A, the units for Gamma Air and Beta Air should not be mSv which is limited to personnel dosage; and what does Total Body dose represent ? (effective dose equivalent ? deep dose equivalent?). Please explain. In Part B, what does Inhalation Doses represent ? (committed dose equivalent? committed effective dose equivalent ?) Please explain. And why more important organs were not chosen such as reproduction gland and red bone marrow, etc. Please explain. In Part D, is the food consumption data the U.S. or Taiwan data , which has a big difference? In summary, the calculation code used in this Table (to calculate T body and Kidney) dose not comply with the requirements of 10CFR Part 20 and ICRP-30.

PSAR Response:

The provided Gamma Air dose represents a skin dose whereas the Total Body dose is a gamma deep dose equivalent total dose.. The final dose assessments made using USNRC RG 1.109 dose conversion coefficients (DCFs), when compared to ICRP-30 dose algorithms, provide doses which are generally comparable and in most cases bounded by the ICRP-30 values. As an example, the following table compares the dose conversion coefficients from RG 1.109 to those found in Federal Guidance Report (FGR) 12 which is the formal standard in the U.S. for assessing compliance to ICRP 30. The table shows that the overall RG 1.109DCFs are more conservative than the FGR 12 values and that in a total body dose calculation it can be expected that the RG 1.109 calculation will bound the ICRP 30.

Comparison of Dose Conversion Factors

	R.G 1.109 Gamma Air	FGR 12 Skin	R.G 1.109 Gamma Total	FGR 12 Total Body
	Gy/yr / Bq/cm3	Sv/yr / Bq/cm^3	Gy/yr / Bq/cm^3	Sv/yr / Bq/cm^3
KR-83m	5.21E-03	1.16E-03	2.04E-05	1.08E-04
KR-85	4.64E-03	4.04E-03	4.35E-03	2.82E-03
KR-85m	3.32E-01	2.73E-01	3.16E-01	1.97E-01
KR-87	1.67E+00	1.58E+00	1.60E+00	1.13E+00
KR-88	4.10E+00	4.00E+00	3.97E+00	2.88E+00
KR-89	4.67E+00	3.60E+00	4.48E+00	2.58E+00
KR-90	4.40E+00	2.42E+00	4.21E+00	1.71E+00
Xe-131m	4.21E-02	2.31E-02	2.47E-02	1.06E-02
Xe-133	9.53E-02	6.72E-02	7.94E-02	4.22E-02
Xe-133m	8.83E-02	5.97E-02	6.78E-02	3.65E-02
Xe-135	5.18E-01	4.32E-01	4.89E-01	3.06E-01
Xe-135m	9.07E-01	7.80E-01	8.42E-01	5.42E-01
Xe-137	4.08E-01	3.43E-01	3.83E-01	2.41E-01
Xe-138	2.49E+00	2.23E+00	2.38E+00	1.59E+00
Ar-41	2.51E+00	2.40E+00	2.39E+00	1.69E+00

The inhalation dose provided in the PSAR is a committed dose equivalent and is not a committed effective dose equivalent since the DCFs used were taken from RG 1.109. The doses for specific organs, such as reproductive, bone marrow etc., were not provided due to the current unavailability of a specific approved computer model for the calculations.. These individual organ dose commitments will, however, be provided in the FSAR in order to demonstrate compliance with the recent version of 10CFR20. Likewise the food consumption rates specific to the area around the Lungmen NPS will be used in the FSAR submittal in order to assess potential normal dose commitments.

PSAR subsection 12.2.2.4 will be changed as follows: After the seventh sentence the following sentence will be added, "The food consumption rates specific to the area around the Lungmen site will be used in the FSAR submittal." After the ninth sentence the following sentence will be added: "The doses for specific organs, such as reproductive, bone marrow etc. will be also supplied in the FSAR."

The units and table 12.2-21, part A will be changed to mGy. The values will be unaffected.

ROCAEC Review Comment:

A comparison table of DCF is provided in the response and it is shown that the R.G. 1.109 DCFs are more conservative than FGR 12 DCFs. How about the other factors?

Further Clarification:

RG 1.109 DCFs, when compared to FGR 12 DCFs, are typically more conservative for all pathways because the USNRC chose to overestimate the radiological impact due to each specific radioisotope in order to preserve a margin of caution when converting concentrations to dose rates. The other factors utilized by the USNRC in determining radiation doses also tend to envelope the values of the FGRs (i.e.the USNRC values tend to be more conservative than their FGR counterparts).

ROCAEC Review Comment:

請比較 R.G. 1.109 及龍門廠址附近居民食用量因子，以及輻射劑量計算結果，以證明目前 PSAR 第 12.3 節之分析值確實較為保守。

Further Clarification:

經參考本公司民國八十年十二月出版之「核能四廠第一、二號機發電計畫環境影響評估報告」，比較 R.G. 1.109 及龍門廠址附近居民食用量因子如附表一，輻射劑量結果如表二。

比較結果顯示，採用 R.G. 1.109 食用量因子分析之廠界總輻射劑量（二部機合計值）確實較為保守。

龍門廠址附近居民生活及飲食習慣，將在龍門核電廠裝填核燃料前兩年重新調查更新。

表一

Table 1: Comparison table of R.G. 1.109 and Lungmen EIA Usage Factors

	R.G. 1.109				Lungmen EIA Report			
	Infant	Child	Teen	Adult	Infant	Child	Teen	Adult
Inhalation(m^3/yr)	1400	3700	8000	8000	1400	3700	8000	8000
Drinking Water (l/yr)	330	510	510	730	330	510	510	730
Fruit, Vegetable, Grain (Kg/yr)	-	520	630	520	-	590	700	710
Leafy Vegetables (Kg/yr)	-	26	42	64	-	33	54	83
Milk (l/yr)	330	330	400	310	36	36	23	27
Meat & Poultry (Kg/yr)	-	41	65	110	-	30	48	82
Fish (Kg/yr)	-	6.9	16	21	-	20	46	60
Other Seafood (Kg/yr)	-	1.7	3.8	5	-	3.8	8.0	10.6
Shoreline Recreation (Hr/yr)	-	14	67	12	-	63	84	315

表二

Table 2: Dose Calculation Results by Using Different Usage Factors

	R.G. 1.109 **Usage Factor (PSAR Ch. 12)**		**Lungmen EIA Usage Factor**	
	mGy/yr/unit	**mSv/yr/unit**	**mGy/yr/unit**	**mSv/yr/unit**
A. Airborne Pathway				
Air gamma	0.013		0.035	
Air beta	0.017		0.03	
Total body		0.012		0.017
Skin		0.027		0.036
Iodine, particulate		0.00038		0.017
B. Liquid Pathway				
Total body		0.024		6.89(-5)
Organ		0.055 (Adult bone)		4.65(-5) (Child bone)
C. Site boundary total (2units)		0.072 (mSv/yr/site)		0.034 (mSv/yr/site)

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-005

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

Table 12.2-23 calculates the internal dose then why annual dose (mSv/yr) was used but not committed dose (mSv)?

PSAR Response:

The doses in the table representing internal doses provide a 50 year dose commitment for a one year exposure from a single site. This is the basis for the mSv/yr value. Such exposures can be read as "χ" mSv committed dose per year of exposure.

The following note will be added to Table 12.2-23: "* The doses in the table representing internal doses provide a 50 year dose commitment for a one year exposure"

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-006

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

In Table 12.2-30, the value for the Totals listed at the lower right hand corner of 3.8E+01 was wrong.

PSAR Response:

The value listed at the lower right hand of Table 12.2-30 will be changed from "3.8E+01" to "2.1E+03".

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-007

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

1. Tables 12.4-1 and 12.4-2 are similar calculations and why don they have same units?
2. The dosage units used in this chapter are quite confusing (Sv, Gy, rem and R have all been used). Sometimes the text and the Table are not consistent such as 12.4 and Table 12.4-1; sometimes the wrong units were used such as the mSv unit was used for Gamma air and Beta air in Table 12.2-21. SI units should be used for all units. Please also explain if there is any special consideration for using the Sv and Gy units.

PSAR Response:

1. Table 12.4-2 will be modified so that the the third column is expressed in μSv/h and the fourth column is expressed in Sv/yr.
2. Table 12.2-21, as regards the units for Gamma air and Beta air will be changed to mGy.. Both ROC-AEC document "Ionizing Radiation Protection Safety Standards" and U.S. Code of Federal Regulations 10CFR20, indicate that it is appropriate to use Sieverts (Sv) for describing the quantity of dose equivalency but the tables in R.G. 1.109 list these values in mrad so that the SI unit should be mGy
   The use of Gy units, i.e., absorbed dose or amount of energy deposited, is typically reserved for quantitative assessment of the effects of radiation to inanimate objects, such as equipment. For personnel protection, the biological effects due to radiation exposure are paramount and therefore the unit of Sv is utilized.
   Table 12.4 will be modified throughout to show occupational exposure and dose rates in μSv/h.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-008

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

Table 12.5-2中Tritium Counting的Qty為多少？

PSAR Response:

Table 12.5.2中所列之Tritium Counting儀器係初步規劃構想，僅提供參考。確實採購之儀器型式及數量將於細部設計階段才能決定。依照龍門計畫整體時程之規劃，龍門核電廠之放射化學及相關實驗室將於民國90年進行細部設計。屆時，台電公司將參考當時市場能供應之最新型儀器，選擇最適用者及恰當之數量。實際採用之儀器將於FSAR中敘述。

Table 12.5-2, 12.5-3, 及12.5-4已參考核一、二廠目前使用設備修正，如附頁。修正後之各Tables，將置入PSAR Amendment中。

Table 12.5-2 Laboratory Equipment

Radiation Detected	Type of Instrument	Detector	Qty	Remarks
(1)Gamma Spectrum Analysis	(1)Computer-based multi-ADC analyzer with automatic quantitative analysis capability	(1)30% relative efficiency pure Ge detector with resolution ≦ 2.1keV for 1333 keV peak		(1)
(2)Alpha-beta contamination	(2)Low background alpha-beta counting system with multi-sample 2" planchet	(2)Gas flow proportional counter with background ≦ 1.5cpm and efficiency ≧ 45% for Sr-Y-90		(2)
(3)Alpha counting (filter paper)	(3)Manual type alpha counter with 4" and 2" planchet	(3)Scintillation detector		(3)
(4)Beta counting (filter paper)	(4)Manual type beta counter with 4" and 2" planchet	(4)Scintillation detector	(4)	(4)
(5)Gross gamma counting (liquid sample)	(5)Preamp-amp SCA-timer counter	(5)2" x 2" well type NaI(T1) detector	(5)	(5)
(6)Tritium Counting	(6)Liquid scintillation counter with background ≦ 10cpm and efficiency ≧ 60% for tritium counting	(6)Liquid scintillator	(6)	(6)

Table 12.5-3 Potential Health Physics Instruments

Instrument	Qty	Radiation Detected	Range	Remark
(1)High Pressure Ion Chamber (HPIC)	(1)	(1) Gamma	(1)0-1 mSv/hr	(1)
(2)Dose-rate Survey Meter	(2)	(2)Gamma	(2)0.1 μ Sv/hr -999mSv/hr	(2)Digital LCD display with audible and visual Alarm
(3)Radiation Survey Meter	(3)	(3)Gamma	(3)0-50 μ Sv/hr	(3)NaI Scintillation Detector
(4)Teletector	(4)	(4)Gamma	(4)0-10 Sv/hr	(4)GM type with 16 feet extension tube
(5)Underwater radiation Monitor	(5)	(5)Gamma	(5)0-1 Sv/hr	(5)With cable length more than 100 ft and with audible and visual alarm
(6)Remote Atea radiation Monitor	(6)	(6)Gamma	(6)0-1 Sv/hr	(6)With cable length more than 100 ft and with audible and visual alarm
(7)Portable Area radiation Monitor	(7)	(7)Gamma	(7)0.01-9.99 mSv/hr	(7)With energy compensated GM Tube and LED Display
(8)Waste Bag Monitor	(8)	(8)Gamma	(8)Lower detection limit 37Bq/Kg	(8)6 NAI Scintillation Detector
(9)Neutron radiation Detector	(9)	(9)Neutron	(9)0-20 mSv/hr	(9)He-3 gas filled detector
(10)Portable High Volume Air Sampler	(10)	(10)N/A	(10)0-35 CFM flow rate	(10)Use 4 inch filter paper
(11)Portable Low Volume Air Sampler	(11)	(11)N/A	(11)20-60 LPM flow rate	(11)Use 4 inch filter paper

Table 12.5-3 Potential Health Physics Instruments (con't)

Instrument	Qty	Radiation Detected	Range	Remark
(1)Alpha scintillation Counter	(1)	(1)Alpha	(1)0-50,000 cpm	(1)4 inch detector (2)2 inch detector
(2)Gamma Scintillation Counter	(2)	(2)Gamm a	(2)0-50 mSv/hr	(3)
(3)Beta Scintillation Counter	(3)	(3)Beta	(3)	(4)4 inch detector (5)2 inch detector
(4)Battery Operated portable Air Sampler	(4)	(4)N/A	(4)1 CFM flow rate	(6)
(5)Continuous Airborne Monitor	(5)	(5)Beta, Gamma	(5)0-100000 CPM	(7)With recorder and adjustable alarm, can use charcoal cartridge
(6)Tool monitor	(6)	(6)Beta, Gamma	(6)	(8)Gas proportional counter with large area detector platform

Table 12.5-4 Personnel Monitoring Instruments

Instrument	Qty	Radiation Detected	Range	Remark
(1)Body Contamination Monitor	(1)	(1)Beta, Gamma	(1)Lower limit of detection 37Bq	(1)Large-area proportional detectors with microprocessor-controlled CRT display and speech processor function
(2)Portal Monitor	(2)	(2)Beta, Gamma	(2)Adjustable sensitivity	(2)Plastic scintillation detector with audible and visual alarm
(3)Dosimetry Computer System	(3)	(3)Gamma	(3)NA	(3)On-line dosimetry control
(4)Electrical Pocket Dosimeter	(4)	(4)Gamma	(4)1 μ Sv ~ 999mSv	(4)Adjustable alarm
(5)Digital Alarm Dosimeter	(5)	(5)Gamma	(5)1 μ Sv~ 999mSv	(5)Loud audible alarm
(6)Whole Body Counter	(6)	(6)Gamma	(6)370Bq	(6)NaI detector with fast scanning function
(7)TLD badge	(7)	(7)Beta, Gamma Neutron	(7)0.1mSv~10Sv	(7)With deep dose and shallow dose detection ability
(8)Contamination Frisker	(8)	(8)Beta, Gamma	(8)0~500,000cpm	(8)portable

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-009

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

Table 12.5-4中Pen Dosimeter數量較多，Alarm Dosimeter數量較少，似不符合當前的發展趨勢？

PSAR Response:

§12.5 Table 12.5-2及Table 12.5-3、Table 12.5-4係參考台電公司核能三廠終期安全分析報告所作之初步規劃，僅提供參考。確實採購之儀器型式及數量將於細部設計階段才能決定。依照龍門計畫整體時程之規劃，龍門核電廠之保健物理及相關設備，包括輻射管制區進出門式偵測器、個人劑量偵檢器等將於民國90年進行細部設計。屆時，台電公司將參考當時市場能供應之最新型儀器，選擇最適用者及恰當之數量。實際採用之儀器將於FSAR中敘述。

Table 12.5-2, 12.5-3, 及12.5-4已參考核一、二廠目前使用設備修正，請參閱對審查問題12-008之答覆說明。修正後之各Tables，將置入PSAR Amendment中。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-010

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

Table 12.5-7及Table 12.5-8中，mr是否應為mR? 可否改用mSv表示？

PSAR Response:

(1) 1. mr 是為 mR 之誤。

(2) 2. 遵照修改。龍門初期安全分析報告中有關輻射度量或劑量之表示，將改採用 SI 單位。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **12-011**

PSAR Sections: **Ch. 12**

Question Date: **February 16, 1998**

PSAR Question:

In PSAR Page 12.3-8, Section 12.3.1.3, six access control zones have been designated, i.e., A, B,...F, for dose rate 0 - 1 mSv/h. Is this consistent with the current practice of TPC? Also, the accessibility requirements have been established, e.g., one hour/wk for Zone E. Does TPC have similar regulations?

PSAR Response:

The radiation zones now in the PSAR are not consistent with the current Health Physics practices. Therefore, the PSAR zone designations will be changed and the zone maps updated to these designations in the FSAR.

Zone	Design Dose Rate mSv/hr (mrem/hr)	Description of Occupancy
A	≤ 0.0025 (0.25)	monitored area, unlimited occupancy
B	≤ 0.005 (0.5)	Monitored area, unlimited access
C	≤ 0.05 (5)	Controlled area, non-posted area
D	> 0.05 (5)	Radiation area (posted), RWP requested
E	> 1.0 (100)	High radiation area (posted), RWP, locked entryway and access control required
F	> 5 Gy/hr (500rad/hr)	Very high radiation area (posted), RWP, locked entryway and access control required

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-012

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

P.12.5-9，Table 12.5-2 Laboratory Equipment可能有錯誤：10%relative efficiency Ge(Li) detector with resolution≦2.1Mev for 1333 keV peak，請再檢討。

PSAR Response:

指正正確，正確的內容應為：

10% relative efficiency Ge(Li) detector with resolution＜2.1keV for 1333keV peak。

核四廠將採用30% relative efficiency pure Ge detector with resolution≦2.1keV for 1333 keV peak儀器做為Gamma Spectrum Analysis之用，不再使用Ge(Li) detector。請參閱對審查問題12-008之答覆說明。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **12-013**

PSAR Sections: **Ch. 12**

Question Date: **February 16, 1998**

PSAR Question:

Where do the source terms that are listed in Tables 12.2-8 to 12.2-12 come from and what is the uncertainty? How each nuclide is classified ? Also, how the Table 12.2-18 was generated for CRD source and what is its uncertainty ? Please explain.

PSAR Response:

The values in Table 12.2-8 through 12.2-12 are calculated based upon the source terms given in subsection 11.1 and assume conservative flows and filter efficiencies for the respective units. These tables are designed to provide bounding values and are to be used in the determination of long term shielding parameters. As such, the tables bound the upper concentrations in source terms and are expected to over predict the actual source terms by factors of from 7 to 100.

Table 12.2-18 on CRD is based upon measurements made on an FMCRD at an operating BWR reactor. Please see question 12-002 for additional information

No changes will be made to the PSAR as a result of the response to this Question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-014

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

The source data referenced in Section 12.2.1.2.7 in page 12.2-6 was not provided.

PSAR Response:

The radioactive sources in the steam are provided in subsection 11.1. A reference will be added to this section stating, "A listing of radioactive sources is provided along with the basis for these sources in Subsection 11.1"..

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **12-015**

PSAR Sections: **Ch. 12**

Question Date: **February 16, 1998**

PSAR Question:

What kind of source is used for the Reactor Startup Source as mentioned in P.12.2-6?

PSAR Response:

Five Californium (Cf- 252) sources will be used for the Reactor Startup Sources.

The following sentence will be added to the beginning of Subsection 12.2.1.2.9.1: "Five Californium (Cf-252) sources will be used for the Reactor Startup Sources."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-016

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

Are the values listed in Table 12.2-20 in page 12.2-60 for one unit or two units? What is the relationship between Max. tech. spec. and annual average? The two values for Sr-89 seem to be abnormal.

PSAR Response:

They are single unit values. See the response to question 12-003 for an explanation of Max. Tech. Spec.

The relationship between annual average releases and Max Tech Spec. can be described by the following example: A plant such as the Lungmen NPS can be expected to operate on a daily basis of 185 to 370 MBq per second release of noble gases to the offgas system (values referenced to a 30 minute decay from the pressure vessel exit nozzle). To calculate the annual average, a value 1.5 times the 370 MBq would be used to bound the fluctuations in normal operations. The plant is designed to operate safely on a continuous basis up to 3700 MBq / second. This is the design standard for radiation protection for the plant. For short periods of time the plant can operate safely above 3,700 MBq/second up to a level of 14,800 MBq/second which is maximum permitted release rate established by NUREG-0800, SRP 11.4, paragraph III 2. b. For the purposes of table 12.2-20, the annual average release is calculated based upon an operating condition of 3,700 MBq/second which is the maximum release rate for continuous operation to be compared to the concentrations listed as Max Tech. Spec.

The primary release for Sr-89 is 182MBq from the turbine building (Table 12.2-19). This assessment is without a doubt conservative as it is derived from early reactor experience from reference 12.2-5 but is still well within the limitations applied under current standards. The values given for the

Max Tech Spec are incorrectly transcribed (both in the Lungmen PSAR and the ABWR SSAR) and will be corrected. As an example, the Max Tech Spec for Sr-89 is 1.1E-11 MBq/ml which is significantly above the estimated release.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-017

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

Do the Part A and C in Table 12.2-21 in page 12.2-64 refer to outside body dose?

PSAR Response:

See Reply to question 12-004.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **12-018**

PSAR Sections: **Ch. 12**

Question Date: **February 16, 1998**

PSAR Question:

In Section 12.3, special cobalt content limits have been considered for the various components of NSSS. Should the same consideration be given to special elements for Reactor Pressure Vessel and biological shield concrete?

PSAR Response:

The need for special considerations in material for cobalt limits concerns an assessment of the combination of the potential radioactive cobalt source and the probability that cobalt in that location will be subject to wear resulting in cobalt contamination into the reactor water. Calculated neutron flux levels on the Lungmen NPS vessel and biological shield wall are significantly lower than other BWRs due to the large water gap between the shroud and vessel wall. Though the cobalt level in the vessel steel is controlled, other characteristics of the steel are limiting in the vessel design since the vessel will not serve as a significant source of contamination. The biological shield is designed in accordance with the specifications of ANSI/ANS-6.4.2-1985, and for Lungmen will be exposed to lower levels of neutron irradiation than past BWRs and consequently, other stress/temperature related factors more properly determine the material properties required.

No changes to the Lungmen NPS PSAR will be made as a result of the response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-019

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

The Radiation Zoning dose rate listed in Section 12.3.13 of page 12.3-8 is not the same as the one listed in Table 12.5-1 of page 12.5-8. Should they be consistent?

PSAR Response:

The zone designations will be made consistent with the designations as shown below.

Zone	Design Dose Rate mSv/hr (mrem/hr)	Description of Occupancy
A	≤ 0.0025 (0.25)	monitored area, unlimited occupancy
B	≤ 0.005 (0.5)	Monitored area, unlimited access
C	≤ 0.05 (5)	Controlled area, non-posted area
D	> 0.05 (5)	Radiation area (posted), RWP requested
E	> 1.0 (100)	High radiation area (posted), RWP, locked entryway and access control required
F	> 5 Gy/hr (500rad/hr)	Very high radiation area (posted), RWP, locked entryway and access control required

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-020

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

The Shielding analysis codes and nuclear data used in Section 12.3.2 are all over 30 years old and most of them never been modified in the last 10 years. It is suggested that in the FSAR the updated codes and data be employed.

PSAR Response:

The primary shielding code used on the Lungmen NPS project is a modified version of the QAD computer code originally written in 1968 by R. Malenfant. This code exists in many versions and is known and used world wide. The QAD code, though old is simple, fast, and provides results of reasonable precision for the tasks required. For deep penetration problems, the DORT (recent upgrade from DOT 4.4 see Table 12.3-1) or MCNP 4a computer codes are used as the problem requires. Scatter calculations are performed by using GGG for simplified geometry or MCNP 4a for more complex situations. Overall the philosophy is to use the computer code which provides the level of precision necessary for the task without providing over complex modeling simulations.

Table 12.3-1 will be modified as follows: (a) The entry for "Computer Code Description" "DOT4.4" will be changed to "DORT". (b) The following computer code and description will be added to the table: "MCNP4a" "A general purpose, continuous energy, generalized geometry, time dependent, coupled neutron-photon-electron Monte Carlo transport code system." In addition, a note at the bottom of the table will be added that states: "The primary shielding code used on the project is a modified version of the QAD computer code. For deep penetration problems, the DORT or MCNP 4a computer codes are used as the problem requires. Scatter calculations are performed by using GGG for simplified geometry or MCNP 4a for more complex situations."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-021

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

It is requested that the Monitoring Instrumentation - radiation monitoring system as discussed in Section 12.3.4 be all on-line networked.

PSAR Response:

All non-safety-related radiation monitors that are contained in the Area Radiation Monitoring System, the Containment Monitoring System and the Process Radiation Monitoring System are on-line networked and are configured to continuously transmit information back to the Plant Computer System (PCS) via the Non-Essential Multiplexing System. In turn, the PCS will be capable of displaying necessary information to the operator.

All safety-related radiation monitors that are contained in the Containment Monitoring System and the Process Radiation Monitoring System are connected through class 1E Remote Multiplexing Units of the Multiplexing System (MUX). The MUX provides a redundant and distributed control and instrumentation data communications network. For the status monitoring, the MUX provides, via a 1E to Non-1E gateway to the PCS, to support the status display and alarm of the radiation monitoring system.

The above two paragraphs will be added to PSAR subsection 12.3.4.(4).

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-022

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

1. The outside body dose calculation at the plant boundary was not explained in Section 12.3.
2. Consideration was not given to the boundary dose due to large radwaste storage facility in Section 12.3.
3. The impact due to direct and skyshine radiation dose at the plant boundary was not evaluated or explained in Section 12.4.

PSAR Response:

1. By outside body dose, it is assumed that the radiation pathway is direct gamma shine and not immersion in gaseous radwaste. At the time the PSAR was written, no data on shielding or design of the turbine complex was available. A complete assessment of the shine dose will be added to the assessments found in 12.2 and a description of the shielding requirements added to 12.3 in the FSAR.
2. It is assumed that by radwaste storage facility what is meant is the onsite drum storage facility and not the Auxiliary Fuel Building.. No information was available at the time the PSAR was written with respect to the design. According to design criteria, the outer area around the onsite drum storage facility will be designed as an unlimited access area. This means that the maximum direct exposure dose rate to any individual standing just 5 cm from the storage building wall will be less than 0.005 mSv/hr. Although it should be noted that the location of the onsite drum storage facility in not decided at this time, it is expected that the distance from the facility to the nearest site boundary will be greater than 400 meters (the distance of the Reactor Building to the site boundary). If it is conservatively assumed that the building outside wall surface exposure rate is 0.005mSv/hr and that the distance from the building wall to the nearest site boundary is 400 meters, then it can be estimated that the yearly individual exposure dose at the site boundary is 0.0014 mSv/yr. The information and detailed calculation will be supplied in the FSAR

3. Skyshine is not a consideration in the analysis of section 12.4 since most work at power is done within shielding walls and areas of the plant directly affected by skyshine are not normally occupied during operations. Likewise, the site boundary dose is not discussed under in-plant operational considerations but is discussed under section 12.2 for normal offsite doses. Please see the reply to item 1 above.

Subsection 12.2.2.6 "Compliance with Nuclear Power Plant Environmental radiation Design Specifications" will be added to PSAR Ch.12 and to incorporate the statement of response item 1 and item 2 above.

ROCACE Review Comment:

The direct and skyshine radiation dose to the site boundary from the Auxiliary Fuel Building and On-site Drum Storage Facility should be considered in subsection 12.3

Further Clarification:

External exposure rates from gamma radiation outside the Auxiliary Fuel Building (AFB) are estimated to be less than 20μSv/y (2 mr/y) at the maximum point on the site boundary, and similar calculation shows that external exposure rates from the On-site Drum Storage Facility are less than 10 μSv/y (1 mr/y). The following new subsections 12.3.2.3(7), 12.3.2.3(8) and 12.3.3.2.5 to describe the above results will be added to Lungmen PSAR:

12.3.2.3

(7) The Auxiliary Fuel Pool is designed to hold approximately 8,700 spent fuel bundles in a water pool of depth 11.9 meters as a single layer of bundles. Owing to the age of the bundles, those isotopes normally associated with radiological significance are not present. The primary concern in handling these bundles is the direct radiation shine and the production of radioactive hot particles from the surface evaporation of water containing these particulate species.

Based upon data from the GE Morris facility, the primary contaminants in the pool water will be Cs-137 and Co-60. The levels of these radioisotopes at Morris are typically less than

37,000 Bq per liter of pool water (1 μCi/liter) and these concentrations are also expected for the Lungmen Auxiliary Building Fuel Pool. Radiation shielding for this building is based upon an ORIGEN 2 evaluation of BWR fuel with a bundle average exposure of 42,400 MWd/mt and a minimum out of reactor time of ten years.
External exposure rates from gamma radiation outside the building are estimated to be less than 20 μSv/y (2 mr/y) at the maximum point on the site boundary. A detailed calculation will be performed and final values will be supplied in the FSAR."

(8) The siting of the On-site Drum Storage facility is around 590 meters from the nearest site boundary. The size of the facility is based on a final storage capacity of 20,000 - 55 gallon drums at the end of 40 years, i.e., 250 drums per unit per year. The structure will be designed to limit the dose rate outside the building to 0.0025mSv/hr. The major source of radiation is Cobalt 60. "According to design criteria, the outer area around the onsite drum storage facility will be designed as an unlimited access area. This means that the maximum direct exposure dose rate to any individual standing just 5 cm from the storage building wall will be less than 0.0025 mSv/hr. The distance from the facility to the nearest site boundary is around 590 meters (the distance of the building center to the site boundary). It is conservatively assumed that the building outside wall surface exposure rate is 0.0025mSv/hr and that the distance from the building wall to the nearest site boundary is 550 meters, then it can be estimated that the yearly external exposure at the site boundary to a member of the public continually present in an unrestricted area will be less than 10μSv/y (1 mr/y). The information and detailed calculation will be supplied in the FSAR."

12.3.3.2.5 Auxiliary Fuel Building

The Auxiliary Fuel Building is designed for the long term storage of spent fuel bundles which have been out of the reactor for a minimum of 15 years (Shielding will be design utilizing fuel 10

years out of the reactor for conservative analysis purposes, See 12.3.2.3(7)).
The building is divided into two sections, a clean area containing electrical and control equipment, and the contaminated area consisting of the fuel pool plus clean up systems and HVAC systems. The contaminated area is serviced by a HEPA filtered HVAC system and is instrumented to alarm in the Main Control Room upon detection of high radiation in the Auxiliary Fuel Building HVAC."

ROCACE Review Comment:

(1) 請說明什麼是"GE Morris Facility"。
(2) 評估 AFB 直接輻射時，計算 Gamma 輻射， Neutron 似乎應一併考慮。
(3) 說明 On-site Drum Storage facility 直接輻射是如何計算的。

Further Clarification:

(1) GE Morris facility is a long term spent fuel facility operated by GE. The facility, which no longer accepts fuel loads, stores slightly over 3,000 bundles of BWR fuel with a minimum out of reactor age greater than ten years.

(2) Neutron flux from the spent fuel is due totally to spontaneous fission neutron and (alpha, n) reactions. Production rates are extremely small and pool surface flux levels will be very small based upon previous experience.

(3) 核能四廠 On-site Drum storage facility 預定位於開關場北側，建築用地規劃約為長寬各 125 公尺，其中心位置距離最近廠界為 590 公尺，設計容量為二萬桶，輻射防護設計準則將確保倉庫外圍直接輻射小於 2.5μSv/hr.廠外輻射劑量分析方面考慮下列輻射來源：

1)貯存庫直接輻射，包括向天輻射。
2)貯存桶自廠房運送至貯存庫過程對廠界之體外直接輻射。

因為核四大型廢料倉庫尚未完成細部設計，因此劑量評估係參考「核一廠二號低放射性廢料倉庫興建安全分析報告（修訂版）」之設計及假設條件估算而得之結果。
基本假設條件

1. 廠房貯存區分為上、下兩層，為長方形之建築。
2. 地下貯存區，每桶表面劑量率>2R/hr（以 50R/hr 計），地上貯存區，每桶表面劑量率≤2R/hr
3. 屏蔽厚度，外牆厚 80 公分，屋頂厚 75 公分。
4. 混凝土密度：2.242 公噸／立方公尺
5. 運送桶數 500 桶／年，每車次運載 20 桶。廠區至倉庫距離 1 公里，車速 10 公里／小時。

經估算得廠界最大體外輻射劑量曝露

拖車運送：　<2.6E-5 mSv/yr. *

直接輻射：　<4.3E-4 mSv/yr. **

向天輻射：　<9.24E-4 mSV/yr. **

合計：　<1.38E-3 mSv/yr.

因係初步估算結果，因此，於 PSAR 中，保守承諾未來大型廢料倉庫設計將小於 0.01mSv/yr.

*　核一廠每年為 113 輛次，核四則為 25 輛次。

**　核一廠倉庫設計容量為七萬桶，距廠界 420 公尺，核四廠廢料倉庫容量為二萬桶，距廠界 590 公尺。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-023

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

§12.5 Table 12.5-2及Table 12.5-3、Table 12.5-4所提資料太老舊，應參考核能二廠目前所使用設備加以修正更新。其中管制區進出門式偵測站未考慮進去。

PSAR Response:

§12.5 Table 12.5-2及Table 12.5-3、Table 12.5-4係參考台電公司核能三廠終期安全分析報告所作之初步規劃，僅提供參考。確實採購之儀器型式及數量將於細部設計階段才能決定。依照龍門計畫整體時程之規劃，龍門核電廠之保健物理及相關設備，包括輻射管制區進出門式偵測器，將於民國90年進行細部設計。屆時，台電公司將參考當時市場能供應之最新型儀器，選擇最適用者及恰當之數量。實際採用之儀器將於FSAR中敘述。

Table 12.5-2, 12.5-3, 及12.5-4已參考核一、二廠目前使用設備修正，請參閱對審查問題12-008之答覆說明。修正後之各Tables, 將置入PSAR Amendment中。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-024

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

The codes and standards used should consider the host country first and also comply with the relevant U.S. regulations. But the Taiwan regulations were not used in this chapter.

PSAR Response:

1. Taiwan regulations are governing and U.S. regulations are used as generic suggestions only. Taiwan regulations will be referenced and U.S. regulations used only for clarification of details where needed.

2. Section 12.1.1.3 will be modified as follows:

   12.1.1.3 **Compliance with the Republic of China regulations, U.S. Regulation 10CFR20 and Regulatory Guides 8.8, 8.10 and 1.8**

   Compliance of the Lungmen NPS design with the Republic of China regulation entitled "Standards for Protection Against Radiation" and U.S. regulation Title 10 of the Code of Federal Regulations, Part 20 (10CFR20) is ensured by the compliance of the design and operation of the facility within the guidelines of U.S. Regulatory Guides 8.8, 8.10 and 1.8.

3. Subsection 12.2.2 will be modified so that the first sentence of the first paragraph reads as follows: "...normal plant operations for compliance with the Republic of China regulation entitled "Standards for Protection Against Radiation", 10CFR20 and 40CFR190."

4. Subsection 12.3.2.1. item (1) will be modified as follows: "Limit the exposure of the general public, plant personnel, contractors, and visitors to levels that are ALARA and within the Republic of China regulation "Standards for Protection Against Radiation" requirements."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-024

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

The codes and standards used should consider the host country first and also comply with the relevant U.S. regulations. But the Taiwan regulations were not used in this chapter.

PSAR Response:

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   Compliance of the Lungmen NPS design with the Republic of China regulation entitled "Standards for Protection Against Radiation" and U.S. regulation Title 10 of the Code of Federal Regulations, Part 20 (10CFR20) is ensured by the compliance of the design and operation of the facility within the guidelines of U.S. Regulatory Guides 8.8, 8.10 and 1.8.

3. Subsection 12.2.2 will be modified so that the first sentence of the first paragraph reads as follows: "...normal plant operations for compliance with the Republic of China regulation entitled "Standards for Protection Against Radiation", 10CFR20 and 40CFR190."

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RESPONSES TO ROC-AEC's PSAR QUESTIONS

ROCAEC Review Comment:

The compliance of R. G. 8.8, R.G. 8.10, and R.G. 1.8 shall be addressed in PSAR subsctions 12.1.1.3.1, 12.1.1.3.2, and 12.1.1.3.3 respectively.

Further Clarification:

PSAR subsections 12.1.1.3.1, 12.1.1.3.2, and 12.1.1.3.3 will be modified as follows:

12.1.1.3.1 COMPLIANCE WITH REGULATORY GUIDE 8.8

Regulatory Guide 8.8 "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As is Reasonably Achievable" provides information relevant to attaining goals and objectives for planning; designing; constructing and operating a Light-Water reactor nuclear power plant to meet the criterion that exposures of plant personnel to radiation during routine operation of the plant will be s Low As is Reasonably Achievable" (ALARA). The goals of the effort to maintain occupational radiation exposures ALARA are:

(1) To maintain the annual dose to individual plant personnel ALARA.

(2) To keep the annual integrated (collective) dose to plant personnel ALARA.

The concept of maintaining occupational radiation exposures ALARA does not embody a specific numerical guideline value in the guide position. Rather, it is a philosophy that reflects specific objectives for radiation dose protection in:

(1) Establishing a program to maintain occupational radiation exposures ALARA. The guidelines of the Republic of China regulation entitled "Standards for Protection Against Radiation" and Taipower document which was approved by ROCAEC "The Safety Guide on the Radiological Protection for Nuclear Operation of Taiwan Power Company; (SGRP)" chapter 9 will be followed to establish the details of

the relevant dose limits of different levels, i.e. the recording level, the investigation level, and the intervention level etc.

(2) Designing facilities and selecting equipment.
(3) Establishing a Health Physics program and procedures.
(4) Making supporting equipment, instrumentation and facilities available.

The guidance of R.G.8.8 will be followed in Lungmen nuclear power plant design, construction and operation. Sections 12.1; 12.2; 12.3; 12.5 address the details of policies, design and operation considerations and Health Physics program for ALARA exposures respectively. The contents regarding the policy: plant design and operation considerations for ALARA will be evaluated and updated as it is necessary in the FSAR preparation stage.

12.1.1.3.2 COMPLIANCE WITH REGULATORY GUIDE 8.10

Regulatory Guide 8.10 "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable" describes a general operating philosophy as a necessary basis for a program of maintaining occupational exposure to radiation as low as reasonably achievable. Both this guide and Regulatory Guide 8.8 deal with the concept of ALARA occupational exposures to radiation.

Two basic conditions are considered necessary in any policy or program for keeping occupational exposures as far below the specified limits as is reasonably achievable.

(1) Management commitment
(2) Vigilance by the Radiation Protection staff.

The guidance of this guide are followed in Lungmen nuclear power plant ALARA policy considerations and Health Physics program. To meet the philosophy of the guide position, Sections 12.1 and 12.5 present the details of policy and Health Physics program for ALARA exposures respectively. It will be evaluated and updated if there is any change in the FSAR preparation stage.

12.1.1.3.3 COMPLIANCE WITH REGULATORY GUIDE 1.8

Regulatory Guide1.8 "Personnel Selection and Training" will be followed for relevant training programs to plant Health Physics (HP) staff. The details of HP training program will be supplied with the FSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-025

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

12.1.1.2 Operation Policies, 12.1.3 Operational consideration, 台電現已有三座核能電廠，與運轉有關之輻射防護政策，及如何將運轉經驗回饋至核四之設計等承諾均應在PSAR中提出，不應延至FSAR。

PSAR Response:

1. 台電公司為了確實做到 ALARA 要求，參考了現行運轉之 BWR 電廠(包括核一、二、及日本 K-6, K-7 等電廠)經驗回饋至龍門核電廠設計，一般性設計原則，必須符合下列條件：

- Adequate shielding should be provided for all radioactive areas, systems and components to reduce direct dose rate through walls, floors and ceilings, as well as to reduce dose rates from scattering and streaming through doorways, mechanical penetrations, heating/ventilation penetrations, and electrical penetrations.
- Design, layout and specification of equipment within shielded areas should be carried out so as to permit ready disconnection , replacement of components requiring maintenance as units, and decontamination of such components prior to maintenance.
- Equipment handling radioactive material should be designed to operate either automatically or by using remote-manual techniques.
- Materials selected for use in equipment and piping handling primary coolant should be chosen to minimize production of Co-60.
- Systems containing radioactive fluids should be designed to minimize formation and trapping of crud deposits. Systems which have the potential for significant radioactive crud deposits should be provided with connections to permit chemical decontamination.

設計基本要求必須做到：

- "Zero leakage" fuel.
- Design and proper installation of water treatment and waste processing systems to allow high quality water chemistry standards to be maintained and liquid radioactive waste to be effectively processed.

- Control of materials selection, particularly elimination of cobalt to the extent possible.
- Adequate space for maintenance and equipment replacement.
- Adequate shielding (temporary or permanent).
- Minimize dead legs and crud traps in piping systems. Provide water flushing for drains and traps which can not be eliminated.
- Design features to control the oxygen content of feedwater and thereby reduce corrosion.
- Correction of recognized design and construction problems causing abnormal levels of testing and maintenance.
- Design provision in components and systems to permit cleaning and chemical decontamination.
- Instrumentation to conduct testing and inspection remotely.
- Applications of robotics for cleanup, maintenance and inspection tasks.
- Plant layout to permit personnel movement through plant without proximity to radiation sources.
- Plant layout to assure radioactive components are adequately shielded from normal work areas and nearby components, to allow maintenance while the system is operating.
- Proper systems for remote handling of contaminated resins and filters including packaging for waste disposal.
- Equipment packaging in modules for rapid disassembly for inspection or maintenance.
- Quick disconnects of service lines for rapid replacement of equipment.
- Heating and ventilation system design to control temperature and humidity in radiation work areas.

2. 遵照指示，台電公司檢討後，擬修改 PSAR，增加下列章節：

12.1.1.2 Operation Policies

The Radiation Protection Manual defines the management commitment to ALARA and designates the station personnel who will implement the program. The plant superintendent has the final responsibility for the ALARA program but delegates the authority to implement the program to the chief of the Health Physics Division (HP Div.) This authority includes the responsibility to prevent unsafe practices and to ensure that radiation exposures are maintained ALARA. The chief of the HP Div. serves as the radiation protection manager of the plant. The health physicists report to the chief of HP Div. And assist him in implementing the ALARA program. They supervise the health physics technicians who perform radiation monitoring and dose calculations and handle the day-to-day operation of the radiation protection program. A more detail discussion of the responsibilities, authorities and qualifications of these key personnel are given in section 3-2 of "The Safety Guide on the Radiological Protection for Nuclear Operations of Taiwan Power Company, (SGRP)" approved by ROCAEC.

The chief of HP Div. is responsible for ensuring that the Taipower employees and contractors are trained in radiation protection procedures in compliance with regulations issued by ROCAEC and SGRP mentioned above and that the procedures are implemented. The chief of HP Div. and the health physicists are responsible for correcting any unsafe practice and for stopping any operation considered to be unsafe. Any unsafe condition that is not within their scope of responsibility shall be reported immediately to the plant superintendent.

Station personnel, whose assignments require it, will be trained in radiation protection procedures and techniques, and will be tested annually. Personnel assigned to Unit 1 will be trained and tested prior to startup of that unit. After initial fuel loading, contractors who work in the controlled area of the plant will be trained and tested in radiation protection procedures to the extent required for the safe performance of their jobs. Construction personnel on Unit 2 will be instructed for their protection in the event of an emergency at Unit 1.

The health physicist and/or the chief of the Health Physics Division will review maintenance, refueling, and radwaste system operating procedures which involve significant radiation exposures to verify adherence to ALARA policy prior to their use. ALARA will also be considered when station procedures or modifications are reviewed or revised by the Station Operation Review Committee (SORC). The frequency of review of established procedures is delineated in the plant administrative procedures. As time permits, the health physicist and/or chief of the Health Physics Division will observe the implementation of selected procedures (for operations with high exposure potentials) to identify situations in which exposures may be reduced.

In addition to reviews by management, all employees are encouraged to submit suggestions relating to radiation protection on the company Radiological Safety Suggestion form. These forms are to be submitted to their immediate supervisor. The supervisor will act immediately on any valid suggestions to correct any unsafe condition within the scope of his responsibility. Problems not in his scope of responsibility will be reported to the health physicist or chief of the Health Physics Division.

The Nuclear Safety Department will conduct periodic audits and reviews of the radiation protection procedures and practices to verify that the radiation protection program is functioning properly.

12.1.3 OPERATIONAL CONSIDERATIONS

In accordance with company policy and consistent with the radiation protection regulations issued by ROCAEC and "The Safety Guide on the Radiological Protection for Nuclear Operations of Taiwan power Company", the radiation exposure of plant personnel will be kept ALARA by means of the health physics program discussed in section 12.5. The radiation protection policies and practices contained therein are initiated through the training program discussed in section 13.2 through the Radiation Protection Manual discussed in subsection 12.1.1, and through plant procedures.

Procedures for radiation-related jobs at an operating advanced boiling water reactor will be written and approved for use at Lungmen Units 1 and 2. These operating procedures must be reviewed for ALARA purposes. If the procedure requires review by the SORC, the plant superintendent who is chairman of the committee and/or the chief of the Health Physics Division who is a member of the committee will be responsible for compliance with the ALARA commitment. If SORC review is not required, the chief of the Health Physics Division and/or the health physicist will review it for ALARA purposes.

ALARA techniques embodied in station procedures, training, and work practices are discussed below, as are the criteria and conditions for their use. These techniques will not be employed if it is determined by the chief of the Health Physics Division that the total dose received may be increased or that the dose reduction may be negligible compared to the effort involved to implement the technique.

From operating experience at other BWRs and the Atomic Industrial Forum National Environmental Studies Project report, it has been determined that a large percentage of exposure at an operating BWR occurs during plant outages from maintenance and inspection activities and not form normal operating activities. This is to be expected as during operation, instrumentation and valves can be operated from outside the shield walls, and operators only have to enter cubicles containing radioactive equipment for short periods of time to check equipment. Maintenance and inspection personnel usually must be in proximity to lines, valves, instruments, or other pieces of equipment, which are radiation sources, in order to perform their job.

12.1.3.1 General ALARA Techniques

Described below are several general ALARA techniques. Further information on ALARA techniques incorporated into procedures is given in section 12.5

- Permanent shielding is used, where possible, with workers behind walls or in low-level radiation areas when not actively working in high radiation areas. Temporary shielding, such as lead sheets draped or strapped over a pipe or concrete blocks stacked around a piece of equipment, is used in some areas. Temporary shielding is used only if the total exposure, which includes exposure received during installation and removal, will be effectively reduced.

- Systems and equipment which are subject to crud buildup, such as reactor water cleanup system, residual heat removal system, liquid radwaste system, various pumps, filters and demineralizers, have been equipped with connections which can be used for flushing the system to eliminate potential hot-spot buildup.
- Prior to performing maintenance work, consideration will be given to flushing and/or chemically decontaminating the system or piece of equipment in order to reduce the crud levels and hence personnel exposure.
- Work involving whole body exposure rates in excess of 1 mSv/hr or removable contamination levels in excess of 0.017MBq/100cm^2 will be preplanned so the job can be performed safely with a minimum of personnel exposure.
- On complex jobs or jobs with exceptionally high radiation levels, dry-run training will be used, and in some cases mock-ups will be used to familiarize the workers with the operations they must perform at the jobsite. These techniques will assist in improving worker efficiency and thus minimize the amount of time spent in the radiation field. Normally these efforts will be documented and the experience used to improve future efforts.
- As much of the work as possible is performed outside of radiation areas. This includes reading instruction manuals or maintenance procedures, adjusting tools or jigs, repairing valve internals and prefabricating components.
- For repair jobs of long duration, consideration will be given to setting up a communications network such as sound powered telephones or closed circuit television to assist supervising personnel in checking on work progress from a lower radiation area.
- Special tools or jigs will be used when their use would permit the job to be performed more efficiently or prevent errors, thus reducing the time spent in a radiation area. Special tools may also be used if their use would increase the distance from the radiation source to the worker, thereby reducing the exposure received. These special tools will be used only if the total exposure, including that received during installation and removal, is significantly reduced.
- Access control points will be established in low-level radiation areas because personnel may spend a significant amount of time in these areas changing protective clothing and respiratory equipment. These access points are set up to limit the spread of contamination to as small an area as possible.
- Protective clothing and respiratory equipment are selected to

minimize the discomfort of workers and increase efficiency so that less time is spent in radiation areas. The protective clothing is prescribed by health physics commensurate with the hazards involved and the requirements cannot be modified by other personnel.

- Contamination containments, i.e., glove bags, poly bottles, tents, etc., are used where practicable to allow personnel to work on highly contaminated equipment while minimizing the spread of contamination during the work.
- Individuals will be instructed to remain in low-level radiation areas as much as possible, consistent with performing their assigned jobs. On certain jobs, detailed maps will be provided with the Radiation Work Permit to clearly delineate areas of high radiation levels to prevent inadvertent entry into such areas and to identify lower-level radiation areas.
- Personnel will be assigned alarm dosimeters to allow determination of accumulated exposure at any time during the job.
- On jobs where the radiation levels are unusually high, besides alarm dosimeter a timekeeper will monitor the total exposure time using a stopwatch or similar device. This will ensure personnel do not exceed the limits on time spent in a radiation field and thereby exceed applicable dose limits.
- On major maintenance jobs in high-level radiation areas, the job preplanning will include man-Sv exposure estimates for the job. At the completion of the work, a debriefing session will be held in an effort to determine how the work could have been completed more efficiently, resulting in less accumulated exposure. This information, together with the procedures used and actual man-rem expended, will be compiled and filed for future reference. All radiation aspects, i.e., radiation, contamination, airborne radioactivity, and personnel contamination (external and internal), will be compiled and filed for future reference during preplanning of similar work situations.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-026

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

In Section 12.2.2 it was said that "However, for compliance to workers..., direct evaluations are not contained in this document." Which document would contain this evaluation ?

PSAR Response:

Families of documents are currently under preparation to describe the basis for and levels of radiation to be found in all major areas of the Lungmen NPS. One of these families will be a room by room description of expected airborne contamination levels. Similarly there is a family of documents describing the room by room radioactive sources and maximum levels of gamma radiation in each room as well as shield wall requirements. For the airborne calculations the primary contributors will be (1) sources of airborne contaminants in each room and (2) the HVAC flow rates in the room. The primary contributor is the leakage rate of water bearing components into each room, the specification of which is described in the as procured description of each pump, valve, etc. Given this information, calculations (see Appendix 12A) will be prepared to assure proper levels within the limitations permitted by regulation and the radiation zone requirements given in the PSAR. Finally, as part of Chapter 12, a table (or tables) combining like areas of the plant will be incorporated into the FSAR to show the as calculated contamination rates.

No changes to the Lungmen NPS PSAR will be made as a result of the response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-027

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

In Section 12.3.1.3 Radiation zoning discussion, it was said that the area with greater than 1 mSv/h will be locked to prevent unauthorized entry. Is there any alarm in the control room or control station that will signal unauthorized entry?

PSAR Response:

Yes, radiation areas greater than 1 mSv/hr require radiation work permits to enter and a locked entry system. Unauthorized entry will alarm in the control room.

No changes to the Lungmen NPS PSAR will be made as a result of the response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-028

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

In Table 12.4-1, (1) in the Reactor Building item, the work referred to in first paragraph of 12.4.1 Drywell Dose (1) which stated "Typical values for BWRs for maintenance of these valves is.... and 5,000 hours of reactor building work" was missing; (2) in the Turbine Building item, how the 1,000 Hours were obtained for the condensate? In the Section 12.4.4(3) discussion, it was only mentioned that the dose rate can be reduced by one half but not the 2,000 hours per year reduced by one half.

PSAR Response:

(1) Both drywell and reactor building work were compiled into the first item under the Drywell. Therefore 4,000 h (drywell) at 135μSv/h and 5,000 h (reactor building) at 36μSv/h are expected to be reduced to 2,000 h (drywell) at 18 μSv/h and 2,200 h (reactor building) at 13μSv/h. The sum of the hours is then 4,200 h at an average dose rate of 15μSv/h which is what is given in Table 12.4-1 for MSIV work.

(2) Work in the condensate area was judged to require 2,000 h work and was reduced to 1,000 h with the introduction of titanium condenser tubes requiring less maintenance. The amount of required work in the condensate area will need to be reconsidered when the final turbine design is established.

Sub-section 12.4-4 (3) will be modified by adding the response above to the current description.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-029

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

12.5.2.2(4)，clean area locker room是設在何處？該陳述是否表示所有人員(包括員工及包商)在進入電廠後均要換下私人衣服，改穿工作服？

PSAR Response:

(1) 1. 本公司龍門核電廠一、二號機組將於各別機組進入輻射管制區之前，輻射管站附近設置"Clean area locker room"。至於確切位置將於細部設計時規劃決定。

(2) 2. 進入輻射管制區之所有工作人員，在進入工作區之前均要換下私人衣服，改穿工作服。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-030

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

12.5.1.2(5)電廠緊急計畫是由HP負責嗎？本節應與現行的台電組織分工一致。

PSAR Response:

1. 核能電廠緊急計畫在電廠中有專設緊急計畫工程師統籌相關業務，並不是由保健物理人員負責。
2. 核電廠所屬各部門均為緊急計畫任務編組之成員，依不同的專長負責不同的工作，各司其職。保健物理人員僅是其中的一環。
3. 根據上述說明，龍門PSAR subsection 12.5.1.2 中有關核電廠保健物理課之功能描述的第(5)項 "Establishing Emergency Plan"，將予以修正為"支援緊急輻射偵測隊及緊急救護去污隊"。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-031

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

請說明如何考量一部機組完工試運轉時，造成另一部建造中機組之施工人員劑量情形？

PSAR Response:

龍門核能電廠預計安裝兩部進步型沸水式核能發電機組，依據施工計畫，當一號機併聯發電後，二號機尚有約一年的施工期間。在這段時間裡，二號機施工地區受到來自一號機之各種輻射，對施工人員造成的平均個人劑量及總年劑量，估算結果如下：

(一)個人平均全身劑量(mSv/yr)：

	氣體排放	直接輻射	向天輻射	合計
室內工作	0.0168	--	--	0.0168
室外工作	0.0527	1.0741	0.1144	1.2412

(二)施工人員總劑量**(man-Sv/yr)

(0.0168*1500+1.2412*200)/1000=0.2734

**： 假設輻射曝露時間為每人平均2500小時/年

施工人數為1700人：平均1500人工作於室內，200人工作於室外。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-032

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

Please explain the reasonability of the values of 2×10^{-6} s/m^3 and 4×10^{-8} m^2, respectively, for the meteorology dispersion coefficient (X/Q) and deposition factor (D/Q) in P.12.2-9.

PSAR Response:

The dispersion values in section 12.2 are the generic values used in the ABWR SSAR and were calculated by using a 74 meter stack at 27. sites totaling over 230,000 hourly observations. The dispersion values were calculated by using the XOQDOQ code (NUREG/CR-2919) with the final dispersion values picked to bound all the 230,000+ values calculated. Preliminary calculations for the Lungmen site using local meteorology (1 year) indicated that the annual average dispersion coefficients were bounded by the above values and that these coefficients are a factor of two or more conservative. Therefore, the values given in the concentration and dose tables are conservative by at least a factor of two. To correctly evaluate the dispersion values at Lungmen a five year period will be used because this is a sea-side site along with the final configuration of buildings, stack locations, stack heights, flow rates, and temperatures. This analysis will be documented in the FSAR.

The second sentence of PSAR Subsection 12.2.2.1.(7) will be changed as follows: "...will be updated to a five year period site specific evaluation in the FSAR."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-033

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

Please explain the reasonability of Dilution Factor = 10 in Table 12.2-23.

PSAR Response:

The dilution factor requires a minimum mixing of the radwaste effluent from the point of release from the plant to any member of the public or food supply. This factor was originally introduced for plants using a discharge canal to provide a minimum canal flow rate in which the effluent could be mixed prior to the flow reaching any member of the public or being introduced to an area used for producing food. For a sea discharge line, the mixing rate will need to be measured or calculated to determine the mixing efficiency in the discharge.The Lungmen NPS cooling water circulation system will be designed as a deep sea submerged discharge, as R.G. 1.109 recommends that 10:1 prompt dilution factor can be used for a submerged discharge scheme.. In the plants licensed by GE, the most difficult are plants located on reservoirs or on low flowing rivers. In those cases, requiring sufficient canal flow to provide a dilution factor of 10, has given sufficient dilution to meet the offsite dose requirements, therefore we expect a factor of ten from a deep water discharge to be sufficient

No changes to the Lungmen NPS PSAR will be made as a result of the response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **12-034**

PSAR Sections: **Ch. 12**

Question Date: **February 16, 1998**

PSAR Question:

Please explain whether the Dose Conversion Factors of Reg. Guide 1.109 in P.12.2-10 are consistent with the current Ionizing Radiation Protection Safety Standards and proper modifications should be introduced if inconsistencies are found.

PSAR Response:

The Dose Conversion Factors (DCFs) presented in USNRC Regulatory Guide 1.109 are consistent with the ROC-AEC "Ionizing Radiation Protection Safety Standards" to the extent that the doses calculated using RG 1.109 will not vary greater than +/-10% from equivalent doses calculated using the ROC-AEC values". On an isotope by isotope basis however, the doses attributable to some isotopes may vary from approximately a factor of three to ten between the two documents with the DCFs in R.G. 1.109 being the more conservative. · When all contributions are summed however,, the overall dose commitment will be within 10%. Based on prior experience, the doses calculated from the Regulatory Guide will bound the summation of the individual isotopic doses using the values presented in the current "Ionizing Radiation Protection Safety Standards". Note that the final analysis to be provided in the FSAR will use the DCF's from current ROC-AEC "Ionizing Radiation Protection Safety Standards".

No changes to the Lungmen NPS PSAR will be made as a result of the response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-035

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

Please explain the calculation model and methodology for the dose rate calculation for the residents outside the plant from the radioactive gas and liquid discharge.

PSAR Response:

Please see the reponse to question 12-004. The calculational model for gaseous discharges used pathway analyses both from gamma/beta shine as well as inhalation and ingestion pathways. The analysis was based upon U. S. Reg Guides 1.109 and 1.111. Detailed information pertaining to this analysis is contained in PSAR references 12.2-5, 6, and 7. Liquid discharges were modeled as a direct release to the ocean with a dilution factor of 10 applied during discharge. Reference 12.2-8 provides the details for the models used, which are based upon U. S. Reg Guide 1.113

No changes to the Lungmen NPS PSAR will be made as a result of the response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **12-036**

PSAR Sections: **Ch. 12**

Question Date: **February 16, 1998**

PSAR Question:

Section 12.3.1.3 : In 10CFR20, a radiation zone with over 500 rad/hr has been designated but why such designation was not found in PSAR?

PSAR Response:

The zone designations are being updated consistent with the designations shown below which include a 500 rad/h zone.

Zone	Design Dose Rate mSv/hr (mrem/hr)	Description of Occupancy
A	≤ 0.0025 (0.25)	monitored area, unlimited occupancy
B	≤ 0.005 (0.5)	Monitored area, unlimited access
C	≤ 0.05 (5)	Controlled area, non-posted area
D	> 0.05 (5)	Radiation area (posted), RWP requested
E	> 1.0 (100)	High radiation area (posted), RWP, locked entryway and access control required
F	> 5 Gy/hr (500rad/hr)	Very high radiation area (posted), RWP, locked entryway and access control required

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-037

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

Section 12.3.3 should include explanation the protective effect to the personnel from the ventilation system design.

PSAR Response:

Subsection 12.3.3 describes the radiation design requirements for the HVAC. Section 12.3.3.1 describes the regulatory basis used and 12.3.3.2 provides some design description for the major HVAC systems. The systems themselves and their operational modes are more fully described in Chapter 9.

No changes to the Lungmen NPS PSAR will be made as a result of the response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-038

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

In the section 12.4 discussion of Dose Assessment, were those reduced values of dose rates from BWRs from calculations or from experience? have they been verified?

PSAR Response:

The reduced dose rate values were obtained primarily from operating plant experience. The data and modeling were provided by radiological engineers and health physics personnel from both utilities and industry sources. Radiation sources were determined from information supplied by Japan, the United States and several European countries in which the individual pieces of equipment which make up the Lungmen NPS design were developed and tested. Basic work tasks were laid out, various alternatives to work scheduling and methods discussed and scrutinized, and suggested changes made to the design during initial efforts. The results given in section 12.4 represents the expected performance as a result of this investigation. It must be pointed out the projected doses are based upon an uncompromising effort with respect to maintaining exact water chemistry standards and corrosion control.

The verification of these values is based on the information received to date on the performance of the operating BWRs. The reported performance on these reactors suggests that the initial predicted values are conservative and that the actual reduction in dose rates are better than originally anticipated.

No changes to the Lungmen NPS PSAR will be made as a result of the response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-039

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

Section 12.1.2 : Lungmen PSAR should consider the ALARA of decommissioning to guarantee the overall ALARA design requirements.

PSAR Response:

A new section, 12.1.2.4 ALARA Considerations for Decommissioning, will be added to the PSAR and will address the decommissioning aspect of the Lungmen NPS.

ROCAEC Review Comment:

Please provide the draft of section 12.1.2.4 or committed when the draft will be completed .

Further Clarification:

A new subsection 12.1.2.4, entitled "Decommissioning", will be added to PSAR amendment. The following topics will be discussed within this new subsection.

1. Considerations
    - a) Current Regulatory Standards
    - b) Future Regulatory Standards
    - c) Past-Experience
    - d) Plant Design
        - (1) Physical Design Standards
        - (2) Materials Selection
        - (3) Prevention of Contamination
    - e) Operational Considerations

2. Decommissioning Activities
   a) Initial Shutdown Activities
      (1) Component Removal
      (2) Decontamination Activities
   b) Intermediate Activities
      (1) Long Term Storage
   c) Finalized Activities
      (1) Major Component Removal
      (2) Site Decontamination

3. ALARA Considerations
   a) Design Considerations
   b) Preliminary Decommissioning Considerations
   c) Decommissioning Activities
   d) Summary
4. Conclusions

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **12-040**

PSAR Sections: **Ch. 12**

Question Date: **February 16, 1998**

PSAR Question:

Please explain whether the resins used in filter/demineralizers of the RWCU system (section 12.3.1.4.1) will be of the regenerative design and the reason for it.

PSAR Response:

The resins used in the RWCU system are not regenerative but are single use, designed to be transferred to the backwash receiving tank at the end of each cleaning cycle (about 25 days). The use of non-regenerative resins minimizes the handling of radioactive material as the system is automated and does not require manual manipulation.

For further reasons, that the use of non-regenerative resins will simplify the system design and is operation and eliminate a major potential source of chemical upsets, it also saves extensive reprocessing of regenerant solution in the liquid waste system.

No changes to the Lungmen NPS PSAR will be made as a result of the response to this question.

ROCAEC Review Comment:

請補充核一、二、三廠以往再生及不再生 RWCU 樹脂之經驗，重點在提供再生及不再生與放射性廢料產生量之關係及相關運轉數據。不再生樹脂亦有其缺點，即增加廢料產量，此類廢料亦不易以水泥固化，因此台電宜綜合考量再生與不再生之優缺點，再依實際運轉數據，做為核四設計之基礎。

Further Clarification:

沸水式核能電廠 RWCU 系統水處理設備 Filter/demin 裝設的目的，係過濾反應器爐槽底部累積之顆粒狀雜質及沈積物，使爐水進一步獲得淨化。其主要設計功能是以過濾為主，離子交換為輔，與 condensate demineralizer 系統以離子交換為主，過濾為輔之水處理設備（Denineralizer 除礦器）功能上並不相同。目前各 BWR 電廠（包括核一、二廠）RWCU 系統中使用之 Filter/Demin 過濾器所採用之預敷介質均為纖維素混合樹脂粉末，使用後屬於高放射性淤泥狀廢料，事實上並無法再生重新使用。核四廠 RWCU 系統水處理設施亦為 Filter/Demin 預敷式過濾器，因此亦不再生使用。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-041

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

對工作人員輻射劑量紀錄如何保存，保存期限未提及。

PSAR Response:

1. 台電公司龍門核電廠之工作人員輻射劑量紀錄，將遵照我國法規“游離輻射防護安全標準”第 58 至 61 條「紀錄保存」，以及行政院原子能委員會於 83 年(83)會輻字第 01645 號函核備之台電公司“核能發電相關設施輻射防護工作守則”第十編「紀錄保存與報告事項」相關規定辦理。

2. 下列之條文將補入 PSAR, subsection 12.5.1.4

   1 “(9)Medical Surveillance program and Individual Dose Record Keeping Program are provided in accordance with the guidance of the Republic of China regulation entitled “Standards for Protection Against Radiation” and the guidelines of “The Safety Guide on the Radiological Protection for Nuclear Operations of Taiwan Power Company; (SGRP)” approved by ROCAEC.”

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-042

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

對輻射工作人員在正常狀況下及異常事故後，如何實施體檢以保障其健康未予提及。

PSAR Response:

1. 台電公司龍門核電廠輻射工作人員在正常狀況下及異常事故後之醫務監護計畫，將遵照我國法規"游離輻射防護安全標準"第 36 至 41 條「醫務監護」，以及行政院原子能委員會於 83 年(83)會輻字第 01645 號函核備之台電公司"核能發電相關設施輻射防護工作守則"第五編「醫務監護」相關規定辦理。

2. 下列之條文將補入 PSAR, subsection 12.5.1.4

   1 "(9)Medical Surveillance program and Individual Dose Record Keeping Program are provided in accordance with the guidance of the Republic of China regulation entitled "Standards for Protection Against Radiation" and the guidelines of "The Safety Guide on the Radiological Protection for Nuclear Operations of Taiwan Power Company; (SGRP)" approved by ROCAEC."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **12-043**

PSAR Sections: **Ch. 12**

Question Date: **February 16, 1998**

PSAR Question:

Please explain whether the Dose Assessment results shown in Section 12.4 are consistent with the relevant rules of nuclear power plant environmental radiation design specifications.

PSAR Response:

Please see the reply to question 12-038. Industry experience has shown that for BWRs with good operating practices, a utility commitment to maintaining tight specifications, and overall proper plant management, that occupational radiation doses occur at the lower end of the exposure spectrum. These same factors also contribute to the reduction of environmental releases. The overall approach is therefore consistent with the need to minimize plant releases and to reduce offsite releases.

The following table shows that normal operation dose assessment results can meet the requirements of the Republic of China regulation "Nuclear Power Plant Environments Radiation design Specifications."

	Lungmen NPP		**ROCAEC Design Specification**	
	mGy/yr/unit	**mSv/yr/unit**	**mGy/yr/unit**	**mSv/yr/unit**
A. Airborne Pathway				
Air gamma	0.013		0.1	
Air beta	0.017		0.2	
Total body		0.012		0.05
Skin		0.027		0.15
Iodine, particulate		0.00038		0.15
B. Liquid Pathway				
Total body		0.024		0.03
Organ		0.055 (Adult bone)		0.1

C. Site boundary total (2units)		0.072 (mSv/yr/site)		0.5 (mSv/yr/site)

A new Subsection 12.2.2.6, entitled "Compliance with Nuclear Power Plant Environmental Radiation Design Specifications" will be added to the PSAR. This subsection will provide the above table as a means to compare ROC-AEC yearly dose rate limitations against calculated Lungmen NPP values.

ROCAEC Review Comment:

The table should be revised to account the further clarification of 12-022.

Further Clarification:

According to the further clarification of ROCAEC review comment track number 12-022, the table is updated as follow:

	Lungmen NPP		**ROCAEC Design Specification**	
	mGy/yr/unit	**mSv/yr/unit**	**mGy/yr/unit**	**mSv/yr/unit**
A. Airborne Pathway				
Air gamma	0.013		0.1	
Air beta	0.017		0.2	
Total body		0.012		0.05
Skin		0.027		0.15
Iodine, particulate		0.00038		0.15
B. Liquid Pathway				
Total body		0.024		0.03
Organ		0.055 (Adult bone)		0.1
C. Direct Shine				
Auxiliary Fuel Building		0.02 (mSv/yr/site)		
On-site Drum Storage Facility		0.01 (mSv/yr/site)		0.05
D. Site boundary total (2units)		0.102 (mSv/yr/site)		0.5 (mSv/yr/site)

Lungmen PSAR will be changed according to the above clarification.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-044

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

請說明第12.5.1節Health Physics program與台電現行作業之異同。

PSAR Response:

台電公司各核能電廠之Health physics Program訂定基礎均遵照我國法規"游離輻射防護安全標準"以及奉行政院原子能委員會於83年核備之台電公司"核能發電相關設施輻射防護工作守則"相關規定制訂及執行。因此，龍門PSAR第12.5.1節所述與核一、二、三廠做法上並無不同。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **12-045**

PSAR Sections: **Ch. 12**

Question Date: **February 16, 1998**

PSAR Question:

P 12.5-2.第三段似多餘，請說明之。

PSAR Response:

P 12.5-2.第三段的確為多餘，將遵照指正自PSAR中刪除

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **12-046**

PSAR Sections: **Ch. 12**

Question Date: **February 16, 1998**

PSAR Question:

Contents of Tables 12.2-10, 12.2-12 and 12.2-17 are not the same as in GE ABWR SSAR. Please explain.

PSAR Response:

The Lungmen PSAR contains the updated versions of those tables developed as part of detailed U.S. ABWR design activities completed after ABWR certification.

The values in Table 12.2-12 were updated to correct a series of typographic errors in the exponents of the values and the symbol for Te-129m.. Table 12.2-10 was modified by including a spiking event used to increase the fission product water inventories. This was done to simulate a depressurization event while the RWCU continued to operate. Likewise, Table 12.2-11 was updated for the same event used on Table 12.2-10.

Table 12.2-17 of the ABWR SSAR is scheduled to be changed to the identical values exhibited in the Lungmen PSAR Table 12.2-17 in order to clarify errors originally contained in the SSAR table.

No changes to the Lungmen NPS PSAR -will be made as a result of the response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 12-047

PSAR Sections: Ch. 12

Question Date: February 16, 1998

PSAR Question:

Please correct the following typo errors :

1. The symbol in first column of Table 12.2-3a, Table 12.2-3c and Table 12.2-4b is wrong. For instance, 8>E>10 does not make sense since energy can not be greater than 8 but less than 10.

2. The values of the "Total" at the lower right hand corner of Tables 12.2-10, 12.2-11 and 12.2-17 are not correct.

3. The D/Q value shown in Page 12.2-9 of Section 12.2.2.1(7) is not consistent with the one in Page 11.3-26 of Section 11.3.9.3. Which one is correct ? Also, the 555 MBq shown in Section 12.2.2.1(5) should be corrected to 555 Mbq/s.

4. In Table 12.4-1, the Subtotal 145 man-Sv/yr is wrong which should be corrected to 0.145 man-Sv/yr.

5. The third paragraph of Section 12.2.4.3, Pertinent design parameters and requirements, mentioned that "The alarm setpoints will....as specified in subsection 12.3.7.2,..." but there is no section 12.3.7.2 to be found.

PSAR Response:

1. Yes, it appears all the symbols have been reversed. This will be corrected.

2. It appears that in the printed versions of the PSAR the right most number is cut off. The values appear correctly in the electronic copies of the PSAR. This will be corrected in the printed versions. The

totals value for Table 12.2-10 is 1.8E+06, for Table 12.2-11 the value is 3.0E+06, and for Table 12.2-17 the value is 2.2E+07. Also note that the value for Co-58 in Table 12.2-11 should read 1.4E+02 and not 1.4E+021. This will also be corrected.

3. The values of D/Q are given as $4x10^{-8}$ m^2 in both paragraphs. Unfortunately one paragraph refers to a maximum (as in maximizing the ground deposition) while the other paragraph refers to a minimum. These values (see question 12-032) were calculated to maximize ground deposition and airborne dispersion. The text in 11.3.9.3 will be revised. The value in section 12.2.2.1(5) should read 555 Mbq/s and will be corrected.

4. The man-Sv/yr subtotal for the Reactor Building will be changed from "145" to "0.145".

5. The first sentence of third paragraph of Section 12.3.4.3 will be modified to read: "The alarm setpoints will be established in the field following the installation of the equipment."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： N-13-001

問題章節(PSAR Section)： Chapter 13

初提日期(Question Date)： 1997.12.17

問題內容(PSAR Question)：

- P. 13. 1-2
  The flow rate in both Creeks A and B, …是指那二條河流，請明確標示。
- 請說明貴公司與 NSSS 及 AE 之運作方式、管理、審查規劃。

問題答覆(Responses)：

1. 原 PSAR Page 13.11-2 內 13.1.1.1.1 之(1)(c) Hydrology 內容，擬略修改使更為清楚：
   原文：
   "The flow rate in both Creeks A and B is very small. However, TPC designed their probable maximum flood(PMF) a 1000 year basis."
   修改為：
   "Both the Yenliao Chi Creek, which is a small creek flowing through the site, and the downstream portion of the Shihting Chi Creek, which is away from the area of Unit 1 and 2, were relocated, so that these creeks will not adversely affect the Yenliao plant site."

2. 請參閱附件 「台電公司與 NSSS 及 BOP A/E 之運作方式管理與審查規則」。

台電公司與NSSS及BOP A/E之運作方式管理與審查規則

台電公司與NSSS及BOP A/E之運作方式：

(一) 設計與設計審查之權責

台電公司龍門計畫係採核反應器系統以設備附帶工程／設計之採購及電廠其他系統(BOP)以設備採購(Components Basis)之組合經營方式，故核反應器部份由主承包商奇異公司負責設計，台電公司及台電公司聘僱之 BOP 顧問(A/E)石偉公司並行設計

審查。電廠其他系統係由石偉公司(或設備提供廠家)負責設計，由台電公司進行審查(設備廠家之設計則由台電公司與 BOP 顧問公司並行設計審查)。

(二) 設計與設計審查之管制

台電公司依據 NSSS 與 BOP A/E 契約，進行設計與設計審查之管制方法如下：

1. 台電公司依計劃時程審查奇異公司與石偉公司所提出之設計文件。
2. 台電公司與石偉公司每月定期舉行 Project Engineering Meeting ，討論工程設計概況與相關技術問題。
3. 台電公司與石偉公司不定期與奇異公司舉行 Project Management Meeting ，討論工程設計概況與相關技術問題。
4. 奇異公司與石偉公司每月提出工作月報供台電公司審查。
5. 台電公司對奇異公司與石偉公司作定期與不定期之稽催與稽查工作。
6. 其他工程管理事項。

工程設計及設計文件審查之規劃

依龍門計畫工程劃分情形，有關工程設計與設計文件審查之分工權責規劃如下：

1. 核反應器系統及其附屬系統之設計由 NSSS 廠家 （ GE 公司）全權負責。

2. 全廠其他週邊系統設計將由顧問公司全權負責。

3. 顧問公司須負責審查 NSSS 廠家與 BOP 廠家之設計及整合 NSSS 與 BOP 系統間之界面設計（作業流程圖一、二）。

4. 詳細之工程設計與設計審查分工則依 NSSS 合約規範書 App.D Project Interface for Engineering and Design Work Scope 以及顧問公司合約規範書 App.J Engineering and Design Work Scope 之規定執行。

5. TPC 對所有設計文件，均將依設計文件重要性分別進行設計審查，其中第一級 （ Level I ）與第二級（ Level II ）文件，以及依 ASME 規定需由業主批准之部份，將進行 100 % 之文件審查，而第三級（ Level III ）與第四級（ Level IV ）文件，將進行選擇性審查。

龍門計畫之設計管理

RESPONSES TO ROC-AEC's PSAR QUESTIONS

(一) 設計管理

1. 設計文件審查作業採行文件分級審查方式，即依文件性質分爲第一～四級（Level I ~ IV）。

   設計文件分級原則：

   A. 第一級設計文件（Level I）

      影響電廠之安全性、操作性、維護性及經濟性程度較高之基本設計文件屬之。
      例如：
      初期安全分析報告（PSAR）、終期安全分析報告（FSAR）、環境影響評估報告（EIA）、主要設計進度表（Engineering Primary Schedule）、設計準則、廠房佈置、系統敘述（System Description）、流程圖（Process Flow Diagram）、管儀圖（P&ID）、單線圖、電廠控制及保護功能／邏輯圖、主要設備規範、重要系統、結構之設計分析、計算書等。

   B. 第二級設計文件（Level II）

      影響電廠之安全性、操作性、維護性及經濟性程度較低之基本設計文件屬之。
      例如：
      邏輯線路圖、管線配置圖、襯版圖（Liner Plate）、次要設備或系統之規範、中階設計進度表（Engineering Intermediate Schedule）、設計分析計算、研究報告等。

   C. 第三級設計文件（Level III）

      一般系統或設備之施工設計文件屬之。
      例如：
      安裝及測試說明書、細部設計進度表（Engineering Detailed Schedule）、樓板平面圖及其剖面圖、照明及通訊系統圖、儀器位置圖、儀器管路圖、接地圖等。

   D. 第四級設計文件（Level IV）

      細部設計施工用設計文件屬之。
      例如：

安裝程序書、接線圖、儀器安裝圖、埋件圖、管線立體圖、容器製造安裝圖、鋼架安裝圖、管節及管架製造安裝圖、鋼筋配置圖、與本級有關之設計、分析及計算之設計文件。

對於第一、二級文件及 ASME 規定須由業主批准部份之文件，台電公司及聘僱之顧問公司將共同進行 100％ 的審查，第三、四級文件則爲視文件內容予以選擇性的審查。

2. NSSS 廠商及 BOP A/E 均於工作開始前即提出其工作範圍內將提供之設計文件清單，內容除必須包括各項文件項目外，尚須註明各文件審查等級，出版時程及文件寄送單位，以確立各單位之工作責任與進度，並縮短文件傳（轉）送時間，提高工作效率。

3. 台電公司內各有關單位將依工程設計文件審查管制作業程序書（ NED-L-3.1.4-T ），落實各項文件之審查及管制工作，以確保工程設計品質。

(二) 設計界面管理

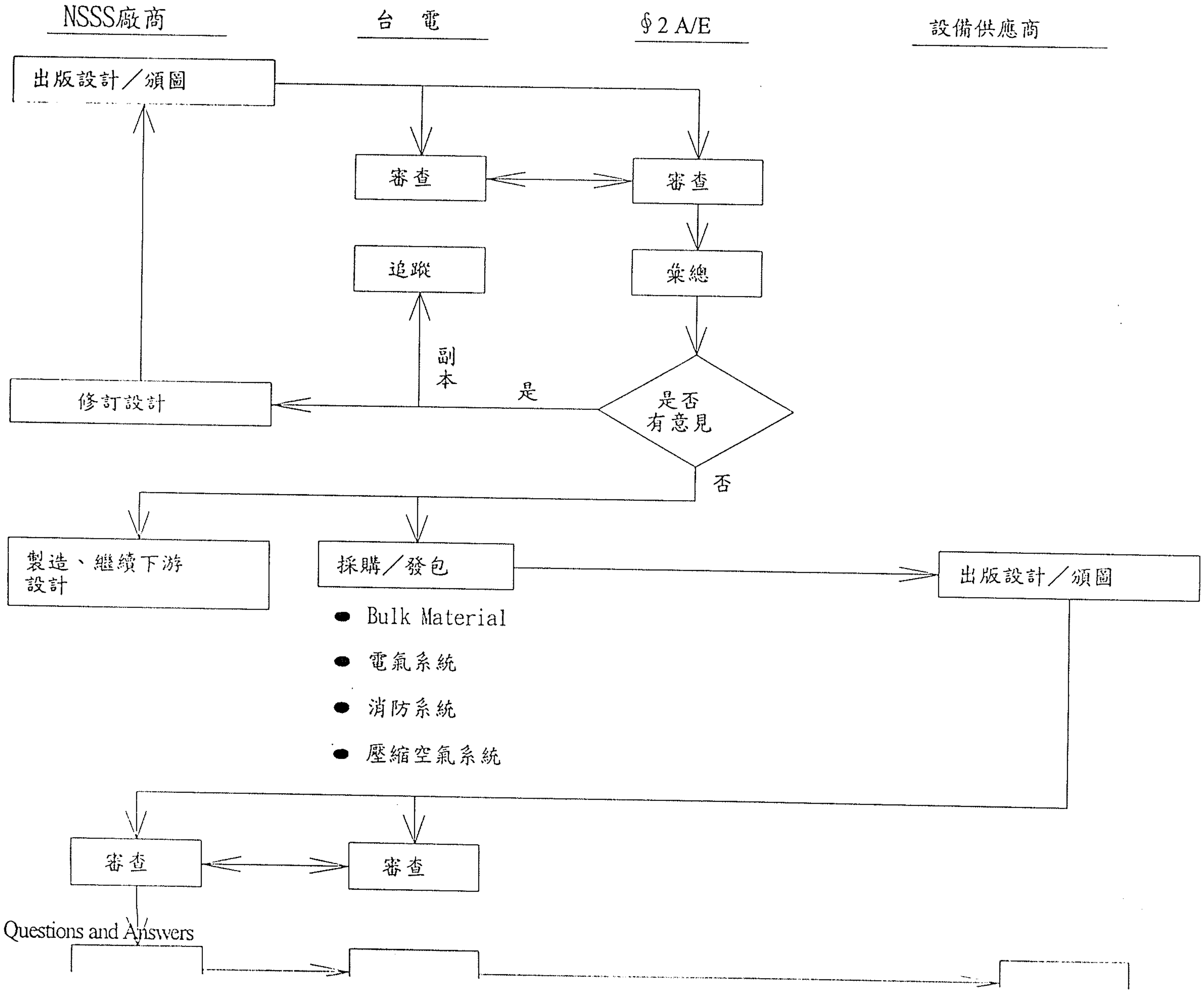
1. 台電公司與 NSSS 合約廠家及 BOP A/E 間之界面，將分別依 NSSS 合約規範書 App.D Table D-2 Project Interface for Engineering and Design Work Scope 及 BOP A/E 合約規範之 App.J Engineering and Design Work Scope 部份之規定辦理。

2. 台電公司單位間之界面，則依各單位權責及核能技術處工程設計文件審查管制作業程序書（ NED-L-3.1.4-T ）之說明辦理。核能技術處爲工程設計文件之主審單位，負責審查、協調及彙總各有關單位依其職掌專長協助審查之意見並函復各原設計單位。

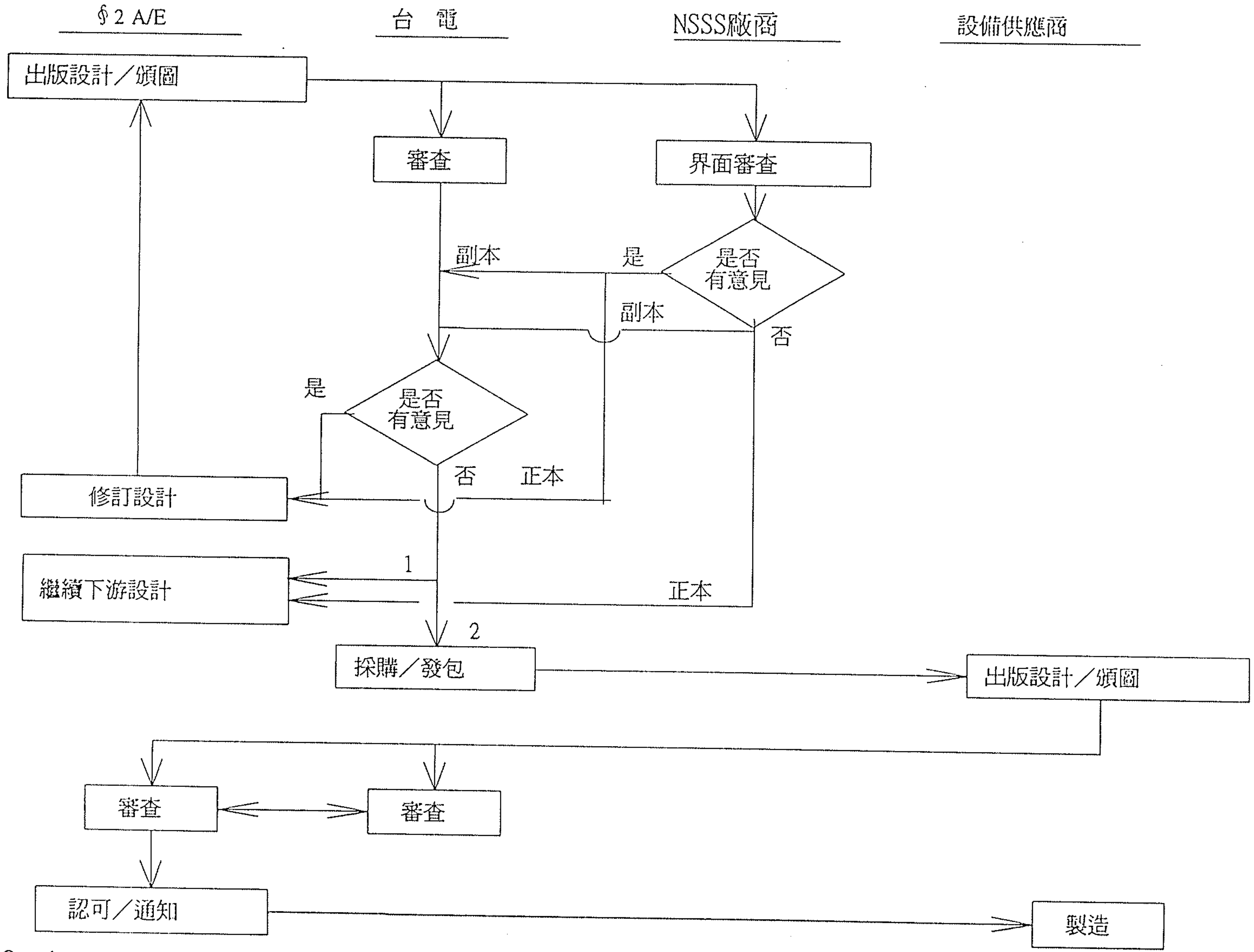
(三) 工程設計文件不合格事項之管制

圖面設計變更如 DCN （ Design / Data Change Notice ）及 FCR （ Field Change Request ）等,比照設計圖發行程序(NED-L-3.1.4-T ）辦理。

圖一、核反應器及其附屬系統設計界面



圖二、其他廠區週邊系統(BOP)設計界面



Questions and Answers

註：1.若需要繼續進行下游設計者

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： 13-002

問題章節(PSAR Section)： Chapter 13

初提日期(Question Date)： 1997.12.05

問題內容(PSAR Question)：

一、 P13.1-6:
Stone & Webster will perform the Design review for all the design documents……, 請說明石偉公司之組織狀況，有關核能審查經驗，人力配置，品質與品保方案執行經驗。

二、建議增加一節： Plant Records

問題答覆(Responses)：

一、(1)石偉公司成立已超過 105 年，在核能工業界的經歷超過 50 年。從參加 1942 年領導開發美國芝加哥大學的人類首座可達到核能自我連鎖反應實驗設備開始，其後持續參與商用核能發電機組，包括第一座商用核能機組 Shipping Port 的設計、建造工作等，總計參與的機組裝置容量達 13000MW 以上。該公司的組織，包括專屬龍門計畫的組織、主要成員的經歷、人力配置等，詳請參閱附件一。
(2)石偉公司的核能審查經驗，參與的美國核能機組超過 17 部，最近幾年有超過百分之九十的美國國內核能機組，都曾經接受該公司提供的技術服務。詳如附件二。
(3)石偉公司的品質與品保方案執行經驗，從參與核能工業，即制訂第一份由美國核管會及其前身美國核准的品保方案 Standard Nuclear Quality Assurance Program(SWSQAP1-74A)，並確實執行。其後並參與 ANSI/ASME NQA-1 的開發工作，且為重要成員。該公司的品保方案執行經驗，詳請參閱附件三。該公司針對龍門計畫使用的品保方案，則在龍門計畫初期安全分析報告第十七章附錄 C 中有完整的說明。

二、基於以下的說明，本公司認為 PSAR 第十七章附錄 A 台電核能工程品保方案，對有關品保活動之記錄已有要求，龍門計畫建造過程所有有關品保的活動，均將依該方案執行，做詳實的記錄及妥善的

記錄保存，建議不需要於 PSAR 中再增加 Plant Records 一節：

(1)根據原能會頒行之「龍門核能電廠執照作業審查基準」，龍門計畫 PSAR 的格式及內容應符合美國法規指引 1.70 第三版之要求。而該版法規指引對第十三章之內容，並無 Plant Records 一節之規定。

(2)美國法規指引 1.70 第一版（1972 年 2 月發行）是曾規定 PSAR 第 13.6 節為 Plant Records，說明如何保留電廠有關品保活動、發生事件之記錄等，但該法規指引於 1972 年 10 月發行第二版時，已取消該項規定。

(3)龍門計畫 PSAR 第十七章附錄 A 台電公司核能工程品質保證方案第十七章「品質保證記錄」也說明「。。。須對影響品質的作業保留充分的記錄作為證據，並建立辦法以管制其品保記錄。。。」，應可確保各項品保記錄的完整。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **13-003**

PSAR Sections: **Ch. 13**

Question Date: **May 4, 1998**

PSAR Question:

1. Figure 13.1-2 on Organization and Functional chart did not show the GE and Stone & Webster positions and their counterpart organizations. This should be clarified.

2. Section 13.6.1 (Preliminary Planning) listed only the U.S. rules and regulations. ROC rules and regulations such as Labor Safety and Sanitation Law and Fire Protection Law, etc. should be included as a minimum.

Response:

1. The chart of the GE's Lungmen Project Team is presented in Figure 1.1 of Document 3113-0A18-0001, which is Attachment 17B to PSAR Chapter 17. The GE Lungmen Project Organization chart was supplied in response to ROC-AEC Question 17-011, Part 1.
   The S&W organization chart is presented in PSAR Chapter 17C, Section 1, Figure 1.
   The attached figure described GE and S&W position and counterpart organizations in the Lungmen project.

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

2. PSAR Section 13.6 (Titled, "Physical Security" per SRP Section 13.6) is supposed to describe "Industrial Security," per Regulatory Guide 1.70, revision 3, or "Protection of Nuclear Power Plants Against Industrial Sabotage" per Regulatory Guide 1.17. Since only the physical security of Lungmen NPS is addressed in this section, it is not appropriate to include requirements for safety and welfare of the work force.

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: **N-14-001**

PSAR Sections: **Section 14.1.1.1.1**

Question Date: **December 3, 1997**

PSAR Question:

The commitments on Post-Construction Tests were just declarations in principle. Please supplement with the following:

1. Please submit post-construction tests plan including procedures list, schedule for execution, list of various systems tests, acceptance criteria and test process management, etc. for the ABWR's new design and special systems.

2. Which department is responsible for flushing operation and how it is carried out thoroughly?

PSAR Response:

1. Lungmen NPS PSAR Chapter 14 was provided in accordance with the guidelines in the RG 1.70, Revision 3. The information provided in the PSAR is not required to address the post-construction test phase. Based on the RG 1.70, PSAR Chapter 14 is to address "Major" phases of the test program, including preoperational tests, initial fuel loading and initial criticality, low-power tests, and power ascension tests. However, a summary description of the post-construction test plan was presented in Lungmen NPS PSAR Section 14.1.1.1.1 to address, in general, how post-construction tests will be performed during the initial test program.

   It is not practical to provide a list of the various system tests and test procedures for the post-construction testing since the post-construction test program covers essentially all of the systems and components throughout the plant for both the NI as well as BOP.

   RG 1.70, Section 14.1.2 specifies that the summary test description should include the test method and test objective in the PSAR, and does not require that the acceptance criteria for those test items described in the PSAR to be included.

As stated in Lungmen NPS PSAR Section 14.1.5, the schedule for post-construction tests is according to the overall construction program. The scheduled time period, relative to the fuel loading date, for conducting the post-construction tests has been presented in the Lungmen NPS PSAR Chapter 14, Figure 14.1-3.

2. With the assistance of the A/E, TPC will provide a test plan for conducting the flushing operation after consulting the NI, T-G and BOP suppliers for the affected systems. As a common practice, the flushing operation is conducted by the Construction Department with support of the plant operating staff.

No change to the Lungmen NPS PSAR will be made as a result of the response to the PSAR questions as stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-14-002

PSAR Sections: Sections 14.1.1.2.1 and 14.1.1.4.2

Question Date: December 3, 1997

PSAR Question:

1. How can one be sure that the Test Director and related personnel will have proper capability to correctly carry out the preoperational and startup tests? Please clarify.

2. Please explain that if design related problems arise during preoperational and startup tests such as design modifications, design deficiencies or errors, and other problems which require retests, how the control and tracking are performed.

PSAR Response:

1. The following will be implemented to ensure the Test Director and related personnel have the capability to correctly carryout the pre-operational and startup test program:

   a. As stated in the Lungmen NPS PSAR Section 14.1.1.2, a sufficient number of qualified people will be assigned to support the preoperational and startup testing from S&W, GE, MHI, TPC and other major equipment suppliers. Qualification and training programs for TPC nuclear plant personnel, including the Lungmen Test Director and related personnel, will be implemented as described in the Lungmen NPS PSAR Sections 13.2 and 13.1.1.1.2.

   b. As stated in the Lungmen NPS PSAR 14.1.1.4, the administrative procedures governing conduct of the initial test program will be contained in the Startup Administrative Manual (SAM). Each Test Director and related personnel will be fully trained and adhere to the program as delineated in the SAM to assure that each preoperatioonal or startup test is correctly performed and the results are satisfactory.

c. As stated in the Lungmen NPS PSAR Section 14.1.1.4.1, all testing is accomplished using approved procedures throughout the preoperational and startup test program. Prior to the test, required personnel are assembled by the Test Director, and the test procedure is reviewed in detail and then performed. During the testing, the plant operating staff is specifically responsible for compliance with operating plant Technical Specifications, compliance with the provisions of the operating license, and authorization of testing.

d. As stated in the Lungmen NPS PSAR Section 14.1.4, information from testing experience acquired with successful and safe startup of ABWRs and over 30 previous BWR plants will be utilized appropriately in the development and implementation of Lungmen startup test procedures.

2. As stated in the Lungmen NPS PSAR Section 14.1.1.5.1, test discrepancies, deficiencies, and omissions identified during testing or during review of test results will be documented as test exceptions. Test exceptions occurring because of design problems will be reported to the appropriate design organization representatives for disposition as stated in the Lungmen NPS PSAR Section 14.1.1.4.2.

   The cognizant design organization will initiate modifications as required. Each exception will be evaluated and assigned a required completion date. These test exceptions are subsequently resolved by processing retesting through the same review and approval cycle as the original testing. Retesting required as a result of a design modification is accomplished using approved procedures and controls in the same manner as described in the Lungmen NPS PSAR Sections 14.1.1.4 and 14.1.1.5.

   Additional information concerning the disposition of design related problems and control and tracking of retests will be developed in the Startup Administrative Manual (SAM) and supplied with the FSAR for ROC-AEC's review when it becomes available.

   No change to the Lungmen NPS PSAR will be made as a result of the response to the above question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **14-003**

PSAR Sections: **Section 14.1.2.1**

Question Date: **December 3, 1997**

PSAR Question:

This part only briefly described the test purpose and test methods which does not meet the RG 1.68 requirements. Please follow the PSAR Table 14.1-1 and list the typical major startup test items (including purpose, prerequisite, test methods and acceptance criteria).

PSAR Response:

As stated above, Lungmen NPS PSAR Chapter 14 was provided in accordance with the guidelines as described in RG 1.70, Revision 3. Based on RG 1.70, Section 14.1.2, summary test descriptions should be included in the PSAR only for those preoperational and/or startup tests planned for each unique or first-of-a-kind principal design features. The summary test descriptions in the PSAR should include the test method and test objectives.

Therefore, no change to the Lungmen NPS PSAR will be made as a result of the response to the PSAR question as stated above.

Further Clarification to ROC-AEC's Comments

Please note that SRP for Section 14.1, "Initial Test Program - Preliminary Safety Analysis Report", has been deleted in accordance with NUREG-0800, Revision 2, dated July 1981. Therefore, it was expected that the SRP for Section 14.2 would only be used to review items relating to the initial test programs described in Chapter 14 of the FSAR. The FSAR will be submitted by TPC as part of the Lungmen NPS operating license application.

The standard format and content specified in Regulatory Guide 1.70 has been used for preparing Lungmen PSAR Chapter 14, Chapter 14.1. The information (including prerequisites and acceptance criteria) requested in this PSAR question will not be developed until the FSAR since it is not applicable for including in the PSAR based on Regulatory Guide 1.70. However, as indicated in Table 14.1-5 of the PSAR, Regulatory Guides 1.68, 1.68.1, 1.68.2 and 1.68.3 are included in the list of Regulatory Guides which will be used in the development of the Initial Test Program for Lungmen NPS.

The information, recommendations and guidance provided in Regulatory Guide 1.68s will also be followed when Section 14.2, "Initial Test Program - Final Safety Analysis Report" is prepared.

The above information should clarify ROC-AEC's comment. No change to the PSAR will be provided as a result of this clarification to the PSAR question as stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **N-14-004**

PSAR Sections: **Section 14.1**

Question Date: **December 3, 1997**

PSAR Question:

Please supplement with detailed description how the following two requirements are met in Chapter 14.

1. SRP 14.2 II.5: "The applicant should incorporate the plant operating, emergency, and surveillance procedures into the test program or otherwise verify these procedures through use to the extent practicable during test program. In addition to verifying the adequacy of plant operating and emergency procedures to the extent practicable during the startup test program, the licensee shall also provide additional operator training during the performance of certain initial tests. This will include training for plant cooldown by means of natural circulation. An acceptance program will satisfy the requirements described in TMI Action Plan Item I.G.1 of NUREG-0660, NUREG-0694 and NUREG-0737."

2. NRC GL 83-24, "Special Low Power Testing and Training" Recommendations for BWRs.

PSAR Response:

1. As stated above, Lungmen PSAR Chapter 14 was provided in accordance with the guidelines as described in RG 1.70. According to NUREG-0800, no SRP is applicable to Section 14.1, Initial Test Program-Preliminary Safety Analysis Report (PSAR) since it has been deleted. The quoted SRP, Section 14.2.II.5, should be used in reviewing the TPC's plans pertaining to the trial use of plant operating and emergency procedures during the initial test program as described in the FSAR Chapter 14 which will be submittal by TPC later as part of Lungmen's operating license (OL) application. Nevertheless, the Lungmen NPS PSAR Section 14.1.6 does describe TPC's plans

pertaining to the trial use of operating and emergency procdures during the period of the initial test program.

Therefore, no change to the Lungmen NPS PSAR will be made as a result of the response to the PSAR question as stated above.

2. As stated above, Lungmen PSAR Chapter 14 was provided in accordance with the guidelines as described in RG 1.70. There is no requirement in RG 1.70 that the PSAR should describe those recommendations which are addressed in USNRC GL 83-24 for special low power testing and training for BWRs. However, this USNRC's recommendation is satisfied by the first two parts of the startup test phase, i.e., 1) initial fuel loading and open vessel testing, and 2) testing during nuclear heatup to rated temperature and pressure condition (approximately 5% power), as described in the Lungmen NPS PSAR Section 14.1.1.1.3,

   Therefore, no change to the Lungmen NPS PSAR will be made as a result of the response to the PSAR question as stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 14-005

PSAR Sections: Section 14.1.1.3

Question Date: December 4, 1997

PSAR Question:

Please explain when Startup Administrative Manual will be submitted?

PSAR Response:

TPC will submit a Startup Administrative Manual (SAM) and any other documents that delineate the following for the ROC-AEC review at the time of the operating license (OL) application: 1) the conduct of the test program, 2) the review, evaluation and approval of test results, 3) the method of controlling prefuel load checks, initial fuel loading, precritical testing and initial criticality, 4) the test program schedule, 5) the specific permissions that are required for the approval of test results and the permission to proceed to the next testing plateau, and 6) the authorization for the determination of operability and availability of interfacing support system requirements.

Therefore, no change to the Lungmen NPS PSAR will be made as a result of the response to the PSAR question as stated above.

<u>Further Clarification to ROC-AEC's Comments:</u>

The generation of a Startup Administrative Manual is not required for PSAR submittal since it is not addressed in the Regulatory Guide 1.70 for Section 14.1. However, the Startup Administrative Manual will be provided in two steps: (1) Step one: one month before the preoperational tests begin for preoperational tests, (2) Step two: two months before the Startup test being for Startup tests..

Therefore, no change to the Lungmen NPS PSAR will be made as a result of this clarification to the PSAR question as stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-14-006

PSAR Sections: Section 14.1.4

Question Date: December 3, 1997

PSAR Question:

The turbine-generators for Lungmen will be supplied by MHI. What is the difference between its startup test and GE or Westinghouse's turbine-generators? Is there any special test requirements on its interface control system with ABWR? Please explain.

PSAR Response:

The startup tests of the MHI turbine-generator consists of: 1).Load Rejection Test, 2) Performance Tests, and 3) Torsional Vibration Test. In general, these tests are not significantly different from either GE or Westinghouse's turbine-generator tests.

The method of performance will be in accordance with the ASME Power Test Code PTC-6 and PTC-6A. The torsional vibration test will confirm that specification criteria for separation of the torsional natural frequencies of the complete turbine-generator shaft from harmonics of line frequencies have been met. The summary test description of the turbine trip and load rejection test was presented in the Lungmen NPS PSAR Section 14.1.2.2.

The turbine trip and generator load rejection test are essentially no different from other BWR plant startup testing. However, the Lungmen NPS is designed to accept a full generator load rejection and/or turbine trip from 100% of rated thermal power or less without reactor trip. In addition, the Lungmen NPS will be able to continue stable operation to supply the house loads after the generator load rejection.

The turbine Electro-Hydraulic Control System (EHC) functional logic and control functions are performed by a triplicated redundant, fault-tolerant digital controller (FTDC). The turbine control system is designed to accept control signals from Steam Bypass and Pressure Control System

(SBPC). In addition, an automatic load demand signal from the automatic power regulator (APR) is provided to the turbine control system for performing the automatic load following (ALF) operations.

Test requirements for the turbine EHC will be developed and supplied with the FSAR submittal after consulting with GE and MHI. The startup test requirements for the SBPC are expected to be essentially no different from the pressure regulator testing performed during the previous BWR plant startups. Additionally, summary test descriptions of plant automation and control startup test and the SBPC preoperational test were presented in the Lungmen NPS PSAR Sections 14.1.2.2.3 and 14.1.2.1.12 respectively.

Therefore, no change to the Lungmen NPS PSAR will be made as a result of the response to the PSAR question as stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 14-007

PSAR Sections: Ch. 14

Question Date: May 13, 1998

PSAR Question:

Is the startup of Lungmen quite different from the existing nuclear power plants? Also, is the operation of the control rods different too? Please clarify.

Response:

The guidelines and recommendations provided by the Regulatory Guide 1.68s are to be used to develop the initial test programs for all nuclear power plants including the Lungmen NPS. Therefore, the Lungmen startup test program is essentially no different from those startup test programs previously performed in other BWR operating plants.

However, the Lungmen initial test program is characterized by some preoperational and startup test testing planned for special, unique, or first-of-a-kind design features of the Lungmen NPS as described in PSAR Section 14.1.2.2. Additionally, the power ascension test portion of the Lungmen startup test program is divided into three sequential testing plateaus (i.e., low-power, mid-power and high-power testing) rather than test conditions (i.e., TC-1 through TC-6) as it was previously performed in other BWR plant startup test programs.

The Lungmen CRD system is composed of three major elements, i.e., electro-hydraulic FMCRD mechanisms, hydraulic control unit assemblies, and CRD hydraulic system. Therefore, the operation of control rods in the Lungmen NPS is not quite the same as other BWR operating plants in Taiwan. The Lungmen CRD is characterized by the FMCRDs which provide electric-motor drive positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid control rod insertion (scram) for abnormal operating conditions. A detailed description of the Lungmen NPS CRD system and FMCRDs is included in PSAR Section 4.6.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **14-008**

PSAR Sections: **14.1.2**

Question Date: **May 10, 1998**

PSAR Question:

It is mentioned in 14.1.2.2.6 that scram trip avoidance margin during the mid power turbine trip test is evaluated and extrapolated to 100% power. How to perform the extrapolation? Please explain.

Response:

The scram avoidance margin for Neutron Flux and Heat Flux is calculated and extrapolated to the high power plateau along 100% rod line by the "rod line ratio", i.e., 100/RL, as follows:

Scram Setpoint - [PO + (100/RL) (Peak Power - Initial Power)]

where

"PO" is the power the plant would be at, if the turbine trip test is to be performed at 100% rod line with the same core flow,

"Peak Power" is the maximum power reached during the transient resulting from mid power turbine trip test, and

"Initial Power" is the power from which the mid power turbine trip test is initiated along the "RL" % rod line.

Note that the methodology of extrapolation described above has been adopted and utilized for several startup test programs at international and US domestic BWR plants.

No changes will be made to the PSAR as a result of the response to the question stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **14-009**

PSAR Sections: **14.1.2**

Question Date: **May 10, 1998**

PSAR Question:

1. Many tests, such as RFCS test and turbine trip, involves verification of SCRRI function. Is it possible to test SCRRI alone? Is it necessary?
2. Is it mentioned in 14.1.2.1.5 that the rod block function of ATLM will be checked. Why the rod block function of RWM & MRBM are not included?

Response:

1. Testing of SCRRI function by itself is not a concern during preoperational phase testing. Please note that the SCRRI function will also be tested as part of the RCIS preoperational test as stated in Section 14.1.2.1.5, Item (2) of Test Method.

   During startup testing, the operability of the SCRRI function could be demonstrated by a manual SCRRI initiation test at the low end of mid power plateau and/or by automatic activation tests in conjunction with a RIPs trip test performed in low enough core flows to assure automatic activation of the SCRRI function for stability control and protection.

   However, such stand alone SCRRI testing during startup phase is not necessary since the operability of the SCRRI function will be verified in conjunction with a major plant transient testing such as turbine trip and load rejection as described in Section 14.1.2.2.6 of PSAR. Adding such testing of SCRRI function during plant operation will result in unnecessary plant transient(s) which is not desirable during plant operation.

   No changes will be made to the PSAR as a result of the response to the question stated above.

2. Please note that as stated in the Test Method, Item (4) of Section 14.1.2.1.5 in PSAR, the RCIS software, including rod worth minimizer (RWM), will be demanded and run to check for correct implementation and operation during preoperational phase testing. Additionally, input signals will be utilized to test the MRBM system input matrix and trip output and verify correct functions in conjunction with the RCIS test during preoperational testing as described in Test Method, Item (10) of Section 14.1.2.1.10 in the PSAR.

No changes will be made to the PSAR as a result of the response to the question stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 14-010

PSAR Sections: 14.1.2

Question Date: May 10, 1998

PSAR Question:

The capability of identifying the presence and location of fault condition about FTDC will be tested in FWCS and SBPC test. Why it is not included in the FTDC test of RFC system? Please explain.

Response:

RFC uses the FTDC in its design the same as FWC and SBPC. Therefore, all the design requirements of FTDC as applicable to the RFC FTDC will be verified during the preoperational phase testing. The capability stated in the question above was inadvertently omitted during the process of generating PSAR Chapter 14 for the Lungmen NPS.

For consistency, the following test item will be added to the end of Section 14.1.2.1.2 in the PSAR as a result of the response to the question stated above:

"(9) Using simulated fault condition(s) to verify that the RFC FTDC is capable of identifying and determining the presence and location of the simulated fault conditions(s)."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **14-011**

PSAR Sections: **14.1.2.1.6**

Question Date: **June 11, 1998**

PSAR Question:

Does the test of EFSAS system include "Response Time Test"? Please Clarify.

Response:

Although it is not specifically addressed in Section 14.1.2.1.6, the response time of ESF actuation system will be included as part of "trip logic check" of the SSLC preoperational test.

As stated in the description of test method, the "SSLC functional logic" from sensor input to driven equipment actuation will be demonstrated through a series of OVERLAP testing. The SSLC logic testing should include test items such as sensor response time tests, logic functional tests and interlock tests as was required to be for non-software based I&C systems during preoperational phase of test program.

No changes will be made to the PSAR as a result of the response to the question stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **I-15-001**

PSAR Sections: **15.1**

Question Date: **November 21, 1997**

PSAR Question:

It happened before in BWR that the feedwater temperature dropped by 66℃ and in PWR, by 111℃ and also power oscillations have occurred as a result of feedwater temperature drop. What ABWR has been different from conventional BWR is that Feedwater Control System (FWCS) will give out warnings when feedwater temperature dropped by 16.7℃ so the operator can lower the power to avoid reactor scram. And the proper functioning of FWCS is therefore very important. If it malfunctions and feedwater temperature continues to drop, it could exceed its set value (55.6℃). Please explain how to make sure the FWCS will function properly and if there are alternatives to alert or monitor the feedwater temperature when it malfunctions.

PSAR Response:

The Feedwater Control (FWC) system provides a feedwater temperature detection scheme, which is single failure proof. The feedwater control system receives individual feedwater line temperature from remote temperature sensors (2 per feedwater line). The two temperature signals per feedwater line are validated with temperature readout provided in the control room. A validation alarm in the control room is presented if the dual temperature signals differ by more than a preset amount for a predetermined period or one signal fails a range limit check. If the individual feedwater line temperatures are valid, they are combined to provide an average feedwater temperature indication in the control room. If the validated individual feedwater line temperatures differ by more than a preset difference, an alarm is presented in the control room. The feedwater temperature signal validity / reasonability checks ensure a high degree of reliability before their use in the FWC system control algorithms.

Feedwater temperature reduction is monitored by the FWC system in two stages (layers). At normal or off-normal conditions, validated average feedwater temperature is continually presented to the control room operator and to the Plant Computer System (PCS) for performance monitoring and,

- For a small decrease in feedwater heating temperature, a loss of feedwater heating alarm is presented. The loss of feedwater heating alarm presented to the control room operator, is developed from validated average feedwater temperature signal compared to an expected feedwater temperature based on main steam flow. The expected feedwater temperature signal, which is a function of steam flow (which in turn is a function of reactor power), is conditioned (lagged) and compared to the validated average measured feedwater temperature to generate the alarm signal.

- For a significant loss of feedwater heating, a Selected Control Rod Run-In (SCCRI) trip is presented. The loss of feedwater heating trip signal which generates a SCCRI, is developed with a similar algorithm as the loss of feedwater heating alarm logic. The SCCRI action will reduce power by an amount necessary to minimize the power transient and protect the fuel from approaching thermal limits.

The consequences of the loss of feedwater heating transient is reduced by implementation of the above control scheme which observes that feedwater heating has been lost, and initiates an action to minimize the power transient.

The loss of feedwater heating transient is expected to be a slow transient caused by closure of the feedwater heater extraction steam shut-off valves, bypass, or isolation of a feedwater heater string. Alarms for equipment failures, feedwater heater valve closures, feedwater heater level, and reactor power increase would be expected to alert the control room operator of a loss of feedwater heating. If the temperature monitoring and control functions are not available, the control room operator takes appropriate actions such as reduction of reactor power by core flow or control rod insertion.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **I-15-002**

PSAR Sections: **15.6.5 : Loss-Coolant Accident-Inside Containment**

Question Date: **November 21, 1997**

PSAR Question:

In Table 15.6-13 of PSAR it was listed that the Thyroid Dose is 1.96 Sv 30 days after LOCA which is different from the 3 Sv listed in SSAR which should be clarified. Also, in the same Table, why the Whole Body dose increases with time ? (the same situation can be found in Table 15.6.14 but a note under that Table indicated it was accumulated values for each of the time periods).

PSAR Response:

The SSAR uses for the Standard ABWR a "maximized" generic meteorology to calculate the offsite 30 day LOCA dose such that the dose would be equal to the 10CFR100 Limitation of 3 Sv (i.e. the meteorology was back calculated to give a 3 Sv dose). For the Lungmen NPS, the meteorology used for calculations was generated based upon a one year base of local Lungmen meteorological data analyzed using the NRC computer code PAVAN (NUREG-/CR-2858). Therefore since the meteorological data is different, the resultant doses are different.

For the FSAR, a minimum of two years of meteorological data is required, therefore some minor adjustment in the meteorological dispersion factors will be seen in the FSAR. In addition, because of the proximity of the turbine building to the site boundary, ground level releases from this pathway will be analyzed for the FSAR using the ARCON96 computer program (NUREG/CR-6331) with a minimum of five years of site meteorological data.

Finally, all the dose evaluations (both thyroid and whole body) given in Chapter 15, as well as environmental releases, are in terms of integrated dose or activity (accumulated values) whereas inventory activities for specific pathways are instantaneous values and are not accumulated. Therefore, all the dose values will be seen to either increase or remain the same. See also the GE response to question track number 15-014 whereby Table 15.6-13 was modified to add a footnote similar to the footnote on Table 15.6-14.

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

ROCAEC Review Comment:

See Track number 15-022.

Further Clarification:

See Track number 15-022.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-15-003

PSAR Sections: 15.6.5 : Loss-Coolant Accident-Inside Containment

Question Date: November 21, 1997

PSAR Question:

Table 15.6-9 to Table 15.6-12 listed the activities of nuclides released to the outside atmosphere during LOCA and the Integrated Activities of various components and in different plant areas. Why some activities of nuclides are lower at the beginning of LOCA but increase after 10 min or even 1h before they finally decay ? Please explain.

PSAR Response:

Reference is made to the reply to question I-15-002. The tables listed in the question are of two types. The first are the activity tables (15.6-9 and 15.6-11) which list the instantaneous isotopic activities in the indicated area/pathway of the plant. The second type of table is the environmental releases (15.6-10 and 15.6-12) which list the isotopic integrated release for dose consequence calculations. At the LOCA initiation only the primary containment will show activity levels and the rest of the pathways will be zero. As the LOCA progresses, the primary containment activity will decrease due to radioactive decay and leakage; and activity levels due to leakage from the primary containment will increase the activity in the follow on volumes until, as a result of radioactive decay and leakage, these volumes peak and finally decrease. Therefore tables of type 1 (15.6-9 and 15.6-11) will show both increases and decreases whereas tables of type 2 (15.6-10 and 15.6-12) will show increases to completion.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-004

PSAR Sections: Ch 15.0

Question Date: January 14, 1998

PSAR Question:

1. According to Table 15.0-1 and Table 4.4-5, Doppler coefficient, void coefficient and core average flow area will not be provided until FSAR. Then how the analysis is accomplished in Chapter 15 ? Please explain.
2. Please explain that in calculating Delta CPR for fast pressurization A00s, why GENESIS Option A procedure (initial condition was set at 102% rated power) was used in SSAR and GEMINI Option A procedure (initial condition set at 100% rated power) was used in Lungmen PSAR ?

PSAR Response:

1. The Chapter 15 transients that were analyzed with Lungmen NPS specific parameters, i.e., GE12 core design and main turbine bypass valve capacity of 110% were;

Event	Analysis Code
Loss of Feedwater Heating	PANACEA
Feedwater Controller Failure - Maximum Demand	ODYNM
Fast Closure of One Turbine Control Valve	ODYNM
Load Rejection Without Bypass	ODYNM
Turbine Trip Without Bypass	ODYNM
Mislocated Fuel Bundle	PANACEA
Misoriented Fuel Bundle	TGBLA

   None of these transients used the REDY transient model for power increase events as shown in Table 15.0-1. Dynamic void, Doppler, and scram reactivity are calculated based on inputs from PANACEA and used as input into the REDY transient model, which consists of a point core model coupled to the recirculation and major system and control and protection models similar to those used for the ODYN transient model. However, the ODYN analysis does not use the Doppler coefficient, void coefficient and core average flow area. ODYN uses more fundamental quantities from the CRNC file. The CRNC file provides axially varying nuclear cross-sections and flow areas.
2. The safety analysis methodology used for analysis of fast pressurization AOOs in Lungmen Chapter 15 PSAR is the GEMINI Option A procedure, whereas the methodology used for ABWR SSAR was the GENESIS Option A procedure. For current BWR analyses or reanalyses, the GEMINI Option A method is used. Under this method, the Technical Specification scram times

given in Table 15.0.5 are used in the safety analysis rather than the statistically evaluated scram times. A statistical adjustment factor, which includes uncertainties of power and methodology was applied in the ΔCPR results of Lungmen specific events in Table 15.0-2.

The ODYN analysis uncertainty for fast pressurization AOOs for Lungmen NPS is given by a statistical adjustment factor (SAFA) which is defined as:

SAFA = (ΔCPR/ICPR)95/95 - (ΔCPR/ICPR)U

where

(ΔCPR/ICPR)U

= Unadjusted licensing analysis ΔCPR/ICPR based on 100% power and the technical specification scram speed.

(ΔCPR/ICPR)95/95

= GEMINI Option A 95/95 ΔCPR/ICPR where the "95/95" denotes 95% probability with 95% confidence that the safety limit will not be violated.

The SAFA value depends on the transient event and include the uncertainties of power and the ODYN model and varies in the range of 0.003 to 0.01 for Lungmen.

Refer to Section 6.3.1.7 of GESTAR III Republic of China, General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-7-RC, August 1995, (Reference 15.7-1) for a description of the GEMINI Option A procedure.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

rack Number: **15-005**

PSAR Sections: **Ch 15.2**

Question Date: **January 5, 1998**

PSAR Question:

1. There is a 150 msec delay for reactor scram and 4 RIPs trip during T/B Trip or Load Rejection condition to confirm if Bypass Valve is open. Please explain when this 150 msec delay starts (stop valve < 85% ? Control valve oil pressure less than setpoint ? T/B Trip initiated signal ? etc.). It was not explained either in Chapter 7.
2. In Table 15.1.7, Reactor scram and 4 RIPs trip inhibited has a 430 msec interval with T/B Trip initiate and a 400 msec interval with Stop valve < 85% open which are different from the 150 msec delay mentioned above. Please explain.

PSAR Response:

1. For the transient analysis load rejection or main turbine trip events, the 150 msec scram delay starts from the stop valve or control valve closure which occurs at time = 0. For the RPS logic, the time delay for reactor scram and 4 RIPs trip starts from the time the load rejection sensing devices trip (control valve oil pressure below the setpoint) to initiate Turbine Control Valve fast closure or the Turbine Stop Valves position switches reach less than 85% open.
2. The original SSAR Table 15.1-7 was modified for the PSAR with additional expected events for Lungmen NPS included. The SSAR assumed a 100 msec main turbine steam shutoff (2.87 seconds into the event, trip signal received and 2.97 seconds after start of the event, stop valves shut). After 2.87 seconds into the event the 150 msec delay should start and the reactor scram and 4 RIPs trip are inhibited due to opening of turbine bypass valves. Table 15.1-7 sequence of events time for reactor scram and 4 RIPs trip are inhibited time will be changed to 3.0 (est.).

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

ROCAEC Review Comments:

1. From the response on the RPS logic design, 150 msec T.D. is counting from when T/B Stop V/V is 85% open but in the analysis in section 15.2.3.3.2.3, 150 msec is counting from when Stop V/V starts to close. From Table 15.2-7 and 15.2-8, it was shown that Stop V/V will be 85% open after 0.01 (sec) so the Scram and 4 RIPs Trip time should be 0.16 sec rather than the 0.15 sec shown in the Table. Please clarify if the analysis which did not consider the time to reach 85% open time is conservative or not.
2. Please clarify during Load Rejection, the Control V/V starts to close either before or after the control oil pressure reaches the RPS setpoint. If the Control

V/V starts to close before the oil pressure reaches the setpoint, then the same time delay question in (1) above should be considered in the analysis.

Further Clarification:

1. For the transient analysis model, the 150 msec time delay scram begins from the start of turbine stop valve or control valve movement to close, i.e., steam flow cutoff to the main turbine commences. The 150 msec time delay represents the maximum analyzed time to initiate inhibit of the RPS logic for scram and RPT. On the other hand, the RPS scram signal and scram inhibit delay timer commence 10 msec (sensor response delay time) after it receives the trip signal from the stop valve limit switches (85% open). The actual RPS scram delay timer operating setpoint will be calculated, based on the combination of bypass valve interrogation versus power level, RPS sensor delay, HCU scram solenoid response (50 ms) and other factors such that the analyzed time of 150 ms is not exceeded. The analysis is conservative, if the total time calculated from start of turbine stop valve closure to the initiation of scram is less than 150 msec.

2. The turbine vendor specifies the decreasing hydraulic trip oil pressure for the control valves fast closure. The RPS control valve fast closure trip functions to provide timely trip signals that are indicative of imminent or actual start of fast closure of the turbine control valves. For the Lungmen design, after receipt of the load rejection signal (Overspeed Protection Controller On), total steam flow through the control valves starts to change after a specific time when control valve hydraulic trip oil pressure decays sufficiently and reaches zero steam flow (control valve full shut) after a specific time based on the characteristics of the control valves. When the control valve fast closure hydraulic trip oil pressure setting is reached (expected to be 4.14 MPa for Lungmen), there is an estimated 10 msec delay before the control valves start to move and steam flow to the turbine starts to decrease. Due to RPS sensor response of 20 msec, the RPS scram signal and scram inhibit delay timer started about 10 msec after the control valves start to actually close to shut off steam flow to the turbine.

   As an example, with control valve trip oil pressure normal, the estimated sequence of events occurs:

	Estimated
Load rejection detected, start of control valve trip oil pressure decay from normal pressure range	time = 0
Control valve trip oil pressure decay to RPS trip setting	50 ms
Control valve closure start with continuing hydraulic oil pressure decay (based on vendors' valve characteristics)	60 ms
Load reject transient start time 0 ms, after control valve starts moving	60 ms

RPS sensor response 20 ms after control valve trip oil pressure setpoint	70 ms
RPS delay timer starts	70 ms

It must be emphasized that the analysis is from start of control valve closure as time zero, and that the RPS timer delay will be calculated based upon consideration of the combination of bypass valve interrogation versus power level, RPS sensor delay, HCU scram solenoid response, and other factors such that the analyzed time of 150 ms is not exceeded.

The PSAR will be revised as indicated in the previous response.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-006

PSAR Sections: Ch 15.4

Question Date: January 14, 1998

PSAR Question:

In Section 15.4.1.2.1 it was mentioned that the probability of a Continuous Control Rod Withdrawal Error during Reactor Startup incident as a "moderate frequency incident" but in SSAR it was termed as "infrequency incident". Does it mean this incident will have a higher frequency of happening in Lungmen than standard design ? Please explain.

PSAR Response:

The TPC Bid Specification Appendix A, Chapter 1: Overall Requirements, Section 2.3.2.3 and Table 1.2-1 item 4.1, required that the Rod Withdrawal Error - Low Power be classified as Moderate Frequency. The NSSS Bid Evaluation: Questionnaire for Clarification, RFPC/GE/S/014 Revision 0, requested that GE clarify the difference of the frequency categories among the Bid Specification (Moderate Frequency), the proposed plant, and the GE SSAR(Infrequent incident). We stated that for the Rod Withdrawal Error - Low Power event, GE believes that this event is Frequency Category "I". However, the results of these events are so mild that GE accepts the Bid Specification classification of MF.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-007

PSAR Sections: Ch 15.6

Question Date: January 14, 1998

PSAR Question:

In the Failure of Small Line Carrying Primary Coolant outside Containment incident, the total amount of the coolant loss to the Reactor Building for Lungmen (13,610 Kg) is about 2.5 times that of the standard design (5442 Kg). Please explain why this difference?

PSAR Response:

SSAR Section 15.6.2.4, second paragraph incorrectly stated that, "The total integrated mass of fluid released into the Reactor Building is 5442 kg." This is corrected in PSAR Section 15.6.2.4. The total integrated mass of fluid released into the Reactor Building is 13,610 kg as correctly shown in Table 15.6-1 of the SSAR and PSAR. Furthermore, in all cases, the environmental releases given in Table 15.6-2 are the same.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-008

PSAR Sections: Ch 15.7

Question Date: January 14, 1998

PSAR Question:

1. In standard ABWR design, Radwaste Building was classified as seismic category I but Lungmen is not. Please explain.

2. In the Fuel-Handling Accident, is it assumed that all the energy from the fuel drop was absorbed by the cladding?

3. In Section 15.7.4.3.3 the words "both impacts" were used but no reference can be found for the sentences before and after these words. Please clarify.

PSAR Response:

1. The radwaste building is designed as seismic category IIB which is accordance with NRC R.G. 1.143, Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Cooled Nuclear Power Plants.

2. The details for determining the number of fuel rods are found in the PSAR Reference 15.7-1 subsection 10.5.1.5. As stated in Reference 15.7-1, one half of the energy is considered to be absorbed by the falling assembly and one half by the four impacted assemblies. No energy is considered to be absorbed by the fuel pellets (i.e., the energy is absorbed entirely by the non-fuel components of the assemblies). The following description will be added to the PSAR Section 15.7.4.3.1 and Section 15.7.4.3.3 with new text shown in italics;

   15.7.4.3.1 Mathematical Model

   The analytical methods and associated assumptions used to evaluate the consequences of this accident are considered to provide a reasonable, yet conservative assessment of the consequences.

   To estimate the expected number of failed fuel rods in each impact, an energy approach is used.

   The fuel assembly is expected to initially impact on the core at a slight angle from the vertical, possibly inducing a bending mode of failure on the fuel rods of the dropped assembly. *It is assumed that each fuel rod resists the imposed bending load by a couple consisting of two equal, opposite concentrated forces. Therefore, fuel rods absorb little energy prior to failure as a result of bending. The energy of the dropped assembly is conservatively assumed to be* *absorbed by only the cladding and other pool structures. Because an unchanneled fuel assembly consists of greater than 70% fuel by weight, the assumption that no energy is absorbed by the fuel material results in considerable conservatism in the mass-energy calculations as described in Reference 15.7-1, Subsection 10.5.1.* After the assembly *initially* impacts the core, the grapple head and mast fall into the core horizontally without contacting the side of the reactor pressure vessel. *The assembly is assumed to tip over and impact horizontally on top of the core from a height of one bundle length, approximately 160 inches.*

15.7.4.3.3 Results

Because of the complex nature of the impact and the resulting damage to fuel assembly components, a rigorous prediction of the number of failed rods is not possible. For this reason, a simplified energy approach was taken and numerous conservative assumptions were made to assure a conservative estimate of the number of failed rods.

The number of failed fuel rods was determined by balancing the energy of the dropped assemblage against the energy required to fail a rod. Using the GE12 10x10 fuel rod array, the analysis resulted in 172 failed fuel rods from both impacts *as described in Subsection 15.7.4.3.1.* See analysis section 10.5.1.5 of Reference 15.7.1.

3. From Reference 15.7.1, the assembly is assumed to impact at a slight angle which is referred to as the initial impact. Then the assembly is "assumed to tip over and impact horizontally on top of the core from a height of one bundle length, approximately 160 inches". This impact from the bundle tipping over is the second impact. See the modifications to Subsections 15.7.4.3.1 and 15.7.4.3.3 above.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-009

PSAR Sections: 15.0 Accident and Analysis

Question Date: November 17, 1997

PSAR Question:

1. The analysis codes ODYNM and REDYA are ABWR versions. Please provide the topical reports of these two codes.

2. What is the difference between ODYNM used in Lungmen and ODYNA used in GE SSAR?

PSAR Response:

1. Topical reports, ODYN Transient Analysis Code NEDC-32083, Revision 0 and REDYA02V Technical Basis Description NEDE-31769P, were provided to Taiwan Power Company under transmittal GETP-1997-0150, March 11, 1997. The ODYN Transient Analysis Code NEDC-32083 covers the ODYNM(10V) and other versions of ODYN as well. The REDYA02V Technical Basis Description NEDE-31769P is the REDYA topical report, which covers other versions of REDY as well. Generally, ODYN (REDY) are the most generic names that relate to a family of codes, ODYNM (REDYA) are more specific to the plant type, and ODYNM10V (REDYA02V) are specific codes.
2. The ODYNA program was originally a special version of the ODYN program designed to simulate the ABWR. Recently, the ODYNA capabilities have been incorporated into the ODYNM program running on the DEC ALPHA workstation. Because of this incorporation, some of the input naming convention used in ODYNA have been changed to make them compatible with ODYNM. The current ODYNM versions contain an additional change, which corrects a coding error which existed in the original ODYNA version. Correction of this error results in a less severe transient response, making the original ODYNA analysis more conservative than those obtained with ODYNM.

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

ROCAEC Review Comments:

1. The report requested in the original question was received and further questions will be asked after reading of the topical report. Here is some preliminary questions about code qualification : the qualification of ODYN included the turbine trip of Peach Bottom and KKM, MSIV closure recirculation pump trip of Hatch and the boron mixing test of Vallecitos. However, no qualification benchmark has been performed for RIP so what guarantee can be achieved for the accurate simulation of the transient behavior of RIP by the ODYN code ?

2. ABWR calculations are based on ODYNA. Since ODYNA analysis results are more conservative, are the qualification conclusions still hold ?

Further Clarification:

1. A qualification of the REDYA computer code for RIPs by comparison with test data from two operating European plants with internal recirculation pumps was performed in the late 1980s. This proprietary, report concluded that the REDYA computer code is acceptable for design analysis for plants with reactor internal pumps. The RIP model used in the REDYA computer code is the same model used in the ODYNA (Lungmen ODYNM) computer code.
2. The ABWR calculations based on ODYNA (Lungmen ODYNM) analysis results are more conservative and the RIP modeling conclusions still hold.

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

ROCAEC Review Further Comments:

1. In your further clarification you mentioned that "A qualification of the READY computer code for RIPs by comparison with test data from two operating European plants with internal recirculation pumps was perform in the late 1980s." Please provide the report or the results of the comparison to demonstrate the RIP simulation capability of READY code.
2. The ABWR calculations based on ODYNA while Lungmen based on ODYNM. The topical report NEDC-32083 shows ODYNA qualification. Since the calculation results of ODYNA are more conservative than ODYNM, it does not necessarily mean that all results are applicable to ODYNM. Please provide sufficient documents or calculation results to demonstrate that ODYNM calculation results are conservative enough.

Further Clarifications:

1. The following report is provided: W. Marquino and R. L. Huang, *Qualification of the REDYA Computer Program: Reactor Internal Pump Test Comparisons*, NEDC-31576P, April 1988, GE Proprietary Information.

2. The ODYN code uncertainty is accounted for in the Statistical Adjustment Factors which are applied as part of the CPR analysis. As stated in the previous response, current ODYNM version contains a change, which corrects a coding error that existed in the original ODYNA version (applicable to ABWR). The original ODYNA code was programmed with redundant calls to subroutines which determine one of the terms in the bulkwater pressurization rate for increasing pressure only. The result was a calculation of an overly rapid pressure increase and overly conservative calculation of the transient Delta CPR and peak vessel pressure. The ODYNM code uncertainty, as represented in the Statistical Adjustment Factors, was adjusted accordingly. The revised ODYNM code

better matched the Peach Bottom Data. Evaluation of this coding change is described in the following GE proprietary information letter, which is provided.

JF Klapproth to the U.S. NRC, *Rectification of Inconsistency in One Dimensional Core Transient Model*, JFK93-50, MFN-176-93, October 29, 1993.

There is no change required to the PSAR from the above further clarifications.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **15-010**

PSAR Sections: **15.1 Decrease in Reactor Coolant Temperature**

Question Date: **November 21, 1997**

PSAR Question:

In the past, BWR has experienced situations where feedwater temperature dropped by 66°C and PWR by 111°C; situations that caused power oscillations due to feedwater temperature drop. Even though the PSAR has indicated that in Lungmen's transient analysis, the feedwater temperature drop of 55.6°C is conservative enough and the ABWR design is different from the conventional BWR in that when feedwater temperature dropped by 16.7°C, the FWCS will give warning to the operator to lower the power output to avoid reactor scram, however, if FWCS fails and feedwater temperature keeps dropping, then it is important whether it will go below the analysis value (55.6°C). So, please explain further the conservatism of the analysis value.

PSAR Response:

The loss of feedwater heating transient analysis assumes that no single operator error or equipment failure shall cause a loss of more than 55.6°C feedwater heating. Based on the reference Heat Balance Diagram Design Flow of PSAR Figure 10.1-2, the worst case number of feedwater heaters which can be tripped or bypassed by a single event, results in a loss of feedwater heating of less than 55.6°C. A feedwater temperature drop of more than 55.6°C is beyond the design basis and the PSAR analysis results are conservative. A loss of 55.6°C feedwater temperature is analyzed to bound this event.

For a multiple failure scenario, a loss of feedwater heating could result in greater than a 55.6°C drop in feedwater temperature. If concurrently a failure of the Feedwater Control System occurs (which has triple redundant controllers), and no high differential actual to reference temperature alarm is received, the failure could go undetected by the operator and reactor scram would occur.

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

ROCAEC Review Comments:

From the GE response, under single failure condition, the feedwater temperature drop should not exceed 55.6 °C. But GE also indicated that under multiple failure condition, feedwater temperature drop can exceed 55.6 °C and if FWCS fails at the same time, reactor will scram. It is also learned from NRC Information Notice 96-41 that Comanche Park PWR has experienced a feedwater temperature drop of 111 °C which was caused by inadvertent open of feedwater heater bypass valve and heater discharge pump trip. So if the same thing happened at ABWR like Comanche Park, should it be regarded as single failure or multiple failure ? Please clarify.

Further Clarification:

For the ABWR design, no single operator error or equipment failure shall cause loss of more than 55 °C (100 °F) feedwater heating capacity. Based on the reference heat balance shown in Lungmen PSAR Figure 10.1-2, the requirement is met as follows:

- isolation of one low pressure heater < 14 °C
- isolation of one low pressure heater string < 38 °C
- isolation of one high pressure heater < 18 °C
- isolation of one high pressure heater string < 32 °C

Heater	ΔT / Heater Stage (°C)	ΔT / Heater (°C)	ΔT / Heater String (°C)
#6	78.28 - 36.81 = 41.47	41.47 / 3 = 13.8	
#5	102.1 - 78.28 = 23.82	23.82 / 3 = 7.94	
#4	132.9 - 102.1 = 30.80	30.80 / 3 = 10.27	
#3	150.7 - 132.9 = 17.80	17.80 / 3 = 5.93	(150.7 - 36.81) / 3 = 37.96
#2	187.4 - 151.7 = 35.70	35.70 / 2 = 17.85	
#1	215.6 - 187.4 = 28.20	28.20 / 2 = 14.10	(215.6 - 151.7) / 2 = 31.95

Therefore, the use of 55 °C (100 °F) temperature drop in the transient analysis is conservative.

For the Comanche Peak Loss of Feedwater Heating (LFWH) event, per NRC IN 96-41, their Chapter 15 licensee FSAR had analyzed for an inadvertent opening of the low-pressure heater bypass valve, coupled with the trip of the heater drain pumps, thus multiple failures. In the actual LFWH event they experienced, extraction steam to the high-pressure heater was lost, resulting in a larger feedwater temperature drop, along with trip of the heater drain pumps.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **15-011**

PSAR Sections: **15.5 Increase in Reactor Coolant Inventory**

Question Date: **November 24, 1997**

PSAR Question:

When HPCF is activated under inappropriate operation, PSAR assumes the HPCF flow rate is 3.2% of the feedwater flow rate and its temperature is a conservative 4.4°C. PSAR then shows the analysis results that the system pressure will be somewhat lower, core neutron flux will be slightly lower than during normal operation. Please evaluate the change in void contents in the core when HPCF was wrongly activated. In theory, when cold water from HPCF is injected into the core, the void contents will decrease which results in neutron flux increase due to positive reactivity addition. Please explain why the analysis results show that the neutron flux actually decreases.

PSAR Response:

As shown in Table 15.0-1a, the HPCF inadvertent startup was analyzed using the REDYA code. The analysis results indicated that void fraction increased by about 0.5%. Operation of HPCF causes a reduction in steam flow which results in a mild reactor depressurization with the pressure regulation system (SBPC) providing corrective action. Cold water injection will reduce void fraction, however, pressure drop will cause void fraction to increase. The flux level increases initially in response to the cold water addition, but settles out at a slightly lower flux level due to the mild depressurization and increase in void fraction.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-012

PSAR Sections: 15.6.5 Loss-of -Coolant Accident - Inside Containment

Question Date: November 24, 1997

PSAR Question:

In Table 15.6-13, the thyroid dose for LPZ is 1.96 Sv (196 rem). In SRP 15.6.5 Appendix A, the statement "At the construction permit (CP) review stage, the staff applies exposure guideline value of 150 rem to the thyroid and 20 rem to the whole body in accordance with Regulatory Guide 1.3 and 1.4." The PSAR result does not conform to SRP guideline value. Please explain.

PSAR Response:

The two step licensing process, i.e., Construction Permit/Operating License, being used for the Lungmen NPS assumes a plant design of low to intermediate detail such that changes in the design could lead to significant performance changes in mitigating accidents. Hence, the two stage process permits construction to begin yet contains margin to allow for design changes. The ABWR design of the Reactor Building and Control Building to be employed at the Lungmen site has been certified in the U.S., and two fully functioning ABWRs are currently in service in Japan. The U.S. ABWR certification dose results also exceed the Construction Permit limits, being 2.4 Sv for the limiting generic meteorology as reported in Table 15.6-13 of the U.S. ABWR SSAR. For the U.S. ABWR Certification combined Construction Permit/Operating License licensing process, the plant design was considered advanced beyond that of the CP licensing stage such that the SRP FSAR limits could be used. Because of completion status of the design, no significant design change leading to a substantial change in the LOCA DBA analysis can occur, and therefore the 3.0 Sv final design limit should be applied.

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

ROCAEC Further Comments:

(1) Please provide practical reasons why the requirements on dose limit during Construction Permit period specified in PSAR 15.6.5 App. A are not applicable to the Lungmen design.

(2) Please explain what the impacts to the Lungmen design will be if dose limit that complies with SRP 15.6.5 App. A is adopted.

Further Clarifications:

SRP 15.6.5 mandates the use of a 150 Rem (1.5 Sv) thyroid dose limit at the CP permit stage to allow for (1) uncertainties in meteorology, and other site related data, and (2) changes in the system design that might influence the final design of engineering safety features on the dose reduction factors of these features. Unlike prior plant projects, the Lungmen NPS design is based upon a U.S. Certified design such that changes in systems or system design is highly unlikely. Those changes specified alone which will change the analysis which are based upon the detailed design are expected to significantly reduce dose levels.

Two major design changes made for the Lungmen NPS which significantly improve the offsite dose response are the use of safety related high SGTS stack and reduction in MSIV leakage rate. The use of stack release significantly improves the dispersion and, therefore, reduces the offsite dose. The MSIV leakage rate for the Lungmen NPS has been reduced to 21.7 L/min. from the value of 66.1 L/min. used for ABWR SSAR. This also leads to a reduction of offsite dose.

The design of main steam system drain line and main condenser in Lungmen plant are dynamic analyzed and the main condenser is bolted to the building basement to prevent walking during an earthquake. For radiological analysis view point, the condenser is sufficiently strong to withstand SSE conditions, it can be modeled as an effective holdup volume and mitigating activities release.

The analysis as presented was made using conservative assumptions such that the doses are maximized for the offsite release. The following factors to the analysis will significantly reduce the offsite dose.

1) The condenser volume is expected to increase from the value used for the PSAR analysis. This will result in increase holdup in the condenser and therefore reduce dose.

2) With the final condenser design, the fraction of the condenser involved in the transport process can be estimated. For existing plants this number approximates 40% of the condenser compared to the 20% currently used in Lungmen PSAR analysis.

   The condenser serves as an effective volume for hold up of all fission products and plateout of non-organic iodine. This is due to the (1) volume of the condenser, (2) surface area, (3) high moisture content which is effective with the highly water soluble salts of iodine, and (4) obstacles to gas transport such as baffle plates. The efficiency of the condenser is conservatively estimated by considering the volume in the condenser above the drain line inlet up the largest point possible rupture or leakage from the condenser. This is typically the turbine seals or other piping interface. For most plants this is equal to approximately 40% of the total condenser volume. For the ABWR SSAR and the Lungmen PSAR this value is set to a more conservative 20% until the turbine condenser design is complete. Upon design completion, the effective volume will be evaluated. Assuming that the Lungmen NPS condenser design falls within the typical range for BWR's of 40%, then the condenser hold up will double resulting in lower release to the environment.

3) The MSIV leak rate will be further reduced by incorporating a calculated LOCA containment pressure of 4 atm, instead of currently used value of 3 atm, in the leakage analysis. Using the new pressure will reduce the leakage rate by approximate 30%.

4) The SGT stack is being relocated and raised to 150 meter to decreased atmospheric depression for elevated release. The increased height will further reduce the dose from elevated release.

The design of system, especially the safety system, for Lungmen has been almost complete. Any design change will be evaluated to ensure that the change will not cause increasing accident offsite dose values and full compliance of Article 8 of Detailed Regulations for Implementation of the Atomic Energy Law of R.O.C. and 10CFR100 of U.S.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-013

PSAR Sections: 15.6.5 Loss-of-Coolant Accident-Inside Containment

Question Date: November 21, 1997

PSAR Question:

Tables 15.6-9 to Table 15.6-12 listed the activities of various nuclides that have been released to the outside during LOCA and the Integrated Activity of various components and plant areas. How come that the integrated activity of some nuclides is lower at the beginning during LOCA but increases after 10 min. or 1 hr before it decays (e.g. the integrated inventory in Table 15.6-9). Please explain.

PSAR Response:

The isotope tables in section 15.6 are of two types. The first are the activity tables (15.6-9 and 15.6-11) which list the instantaneous isotopic activities in the indicated area/pathway of the plant. The second type of table is the environmental releases (15.6-10 and 15.6-12) which list the isotopic integrated_release for dose consequence calculations. At the LOCA initiation only the primary containment will show activity levels and the rest of the pathways will be zero. As the LOCA progresses, the primary containment activity will decrease due to radioactive decay and leakage; and activity levels due to leakage from the primary containment will increase the activity in the follow on volumes until, as a result of radioactive decay and leakage, these volumes peak and finally decrease. Therefore tables of type 1 (15.6-9 and 15.6-11) will show both increases and decreases whereas tables of type 2 (15.6-10 and 15.6-12) will show increases to completion.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-014

PSAR Sections: 15.6.5 Loss-of-Coolant Accident-Inside Containment

Question Date: November 21, 1997

PSAR Question:

The Thyroid Dose after 30 days of a LOCA was listed at 2.4 Sv in SSAR and this value has been improved (dropped to 1.96 Sv) in Table 15.6-13 in PSAR. Please explain the reasons. Also, does the Whole Body Dose in the same Table designate the integrated dose for that time period? If yes, it should be noted below the Table.

PSAR Response:

The SSAR uses a "maximized" generic meteorology to calculate the offsite 30 day LOCA dose such that the dose would be equal to the 10CFR100 Limitation of 3 Sv (i.e. the meteorology was back calculated to give a 3 Sv dose). For the Lungmen NPS calculation the meteorology used was generated based upon a one year base of local Lungmen meteorological data analyzed using the US NRC computer code PAVAN (NUREG/CR-2858). Therefore since the meteorological data is different, the resultant doses are different.

For the FSAR, a minimum of two years of meteorological data is required, therefore some minor adjustment in the meteorological dispersion factors will be seen in the FSAR. In addition, because of the proximity of the turbine building to the site boundary, ground level releases from this pathway will be analyzed for the FSAR using the ARCON96 computer program (NUREG/CR-6331) with a minimum of five years of site meteorological data.

With respect to the final comment, a footnote to Table 15.6-13 to state that both thyroid and whole body doses are integrated doses in a similar manner to the note found on Table 15.6-14 will be added.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-015

PSAR Sections: **15.4.7 Mislocated Bundle Accident**

Question Date: **October 23, 1997**

PSAR Question:

Please describe the differences between the present and previous methodologies for the mislocated bundle accident. Why does the accident become limiting event right now?

PSAR Response:

The GE12 core design is designed with higher radial peaking than the core designs previously analyzed. This core loading has locations which can be driven to a very high power if a reactive bundle is inadvertently loaded to a location surrounded by reactive high power bundles.

The mislocated bundle accident is analyzed against the MCPR safety limit, even though the probability of this event occurring is less than that of most events classified as accidents. Also, this event will not result in 0.1% of the total fuel rods subjected to boiling transition. The USNRC is evaluating this event for reclassification as accident. When this approval comes, it is hoped that the ROC AEC will also reclassify the event, thereby not having it be limiting on Lungmen.

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

ROCAEC Review Comments:

Not accepted for the moment.

1. There is no clarification on whether there is change to the mislocated bundle accident methodology.

2. Is the current radial peaking factor the maximum value that can be tolerated ? Will there be even larger radial peaking factor in the future design ?

3. It is a separate issue and therefore it is suggested that TPC make separate application as necessary to reclassify this incident.

Further Clarification:

1. The current mislocated analysis procedure does not take credit for the monitoring systems ability to detect the mislocated bundle. The delta CPR due

to this methodology difference is less important than the core design described in the previous response.

2. The PSAR mislocated bundle result is based on the Lungmen equilibrium core design. The radial peaking factor for this core is not the maximum value that can be tolerated. It is possible that the ran the core design described in the previous response.

3. Agree

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-016

PSAR Sections: 15.4.8 Misoriented Fuel Bundle Accident

Question Date: October 23, 1997

PSAR Question:

The statement in Section 15.4.8.3.1 says that, "The rotated bundle gap widths assume the bundle is tilted". How is the rotated bundle tilted? How are the gap widths estimated? Please provide a more detailed description on the cause and effect of the accident.

PSAR Response:

The hardware structure at the top of the fuel bundle (channel clip, buttons on the channel, etc.) is configured such that when a bundle is correctly placed in the core, the bundle radial geometry is essentially uniform from the bottom to the top of the bundle. The bundle radial geometry includes the bundle to bundle water gap dimension. When the bundle is rotated, the hardware at the top of the bundle causes the bundle to be tilted, resulting in non uniform radial geometry.

General Electric analyzes the rotated fuel bundle accident with the actual bundle radial geometry, including the axially varying water gap geometry. The goal of the analysis is to determine the change in the Critical Power Ratio (CPR) between the non rotated configuration and the rotated and tilted configuration. A design goal is to assure that the delta CPR is less than the delta CPR set by the limiting transient.

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

ROCAEC Review Comments:

Not accepted for the moment.

1. Please explain further the reason why the misoriented fuel bundle has become one of the MCPR limiting events and clarify if the gap width is the major factor.

2. Please explain in detail with figures how the water gap width tilted during incident and provide actual calculation data according to the steps in the procedure.

Further Clarification:

1. The enrichment distribution for the bundle in question was designed without considering the consequences of the misoriented fuel bundle event. Usually, the enrichment in the fuel pin located in the corner position is reduced to accommodate changes in the water gap thickness caused by bundle misorientation. This problem can be solved at the final bundle design stage by adjusting the enrichment in the corner fuel pin. The gap width and enrichment of the corner fuel pin are the major factors in the resulting ΔCPR of this event.

2. The fuel bundle when misoriented (rotated) and installed in the core results in the fuel bundle being tilted. In a normal arrangement, with four fuel bundles installed in a cell about a control rod, the fuel channel fasteners are oriented centrally towards the control rod, along with the fuel channel spacers which are aligned toward the center. In this normal fuel installation the channel spacers (on two sides of the channel adjacent to the channel fastener) touch and correctly space the four fuel assemblies in a fuel cell. When a fuel bundle is rotated 90° or 180°, the misoriented bundle tilts because the channel spacers no longer provides the proper fuel bundle separation from the other three fuel assemblies.

   PSAR Reference 15.4-3, R. E. Engel (GE) to D. G. Eisenhut (NRC), Fuel Assembly Loading Error, MFN-457-77, November 30, 1977, provides the analysis methods for the misoriented fuel bundle. Table 4-1 in the section on responses to NRC questions of Reference 15.4-3, shows water gap dimensions for different nominal and rotated fuel bundles. In this same reference, NRC Question 6, states, "Provide a discussion of the analysis technique used to ensure that the maximum CPR misoriented bundle is determined. For example, how are variations in water gap difference used to demonstrate that the worst rotation case is chose?" In our response to the NRC question, we discuss calculation step sequences (a through h) used to ensure that the maximum change in critical power ratio in the misoriented fuel bundle has been determined. We also state in the response that since the fuel assembly is rotated and exhibits a slight axial tilting, the water gap sizes for each of these nodes are different. Lungmen specific actual calculation data will be provided shortly.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-017

PSAR Sections: **15.6.2 Failure of Small Line Carrying Primary Coolant Outside Containment**
15.6.4 Steam System Piping Break Outside Containment
15.7.4 Fuel Handling Accident
15.7.5 Spent Fuel Cask Drop Accident

Question Date: **December 12, 1997**

PSAR Question:

Please explain if the dose model, dose conversion factors and breath rates used to calculate the thyroid dose and whole body dose at the outside boundary of EAB/LPZ during accident release comply with the current "Ionizing Radiation Protection Safety Standards".

PSAR Response:

We have reviewed "Ionizing Radiation Protection Safety Standards" and find that the dose factors and limits applied in that document are directed toward annual limits on exposure and intake and not on accident exposure.

The dose factors used for thyroid in the Lungmen PSAR are based upon ICRP 30 dose conversion factors as reported in Federal Guidance Report 11 (PSAR Reference 15.6-6). For the whole body dose conversion factors, a semi-infinite cloud model using the equation 0.25 E_γ where E_γ is the mean isotopic gamma energy in MeV is used (PSAR Reference 15.6-2). The breathing rates are those specified in U.S. Regulatory Guide 1.3.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **15-018**

PSAR Sections: **15.0 Accident and Analysis**

Question Date: **January 6, 1998**

PSAR Question:

In Table 15.0-2 subsection 15.1.1, it is found that the maximum core average surface heat flux and the maximum neutron flux are 126.4% of initial and 115.3% NBR respectively. On the other hand, the result of the same event in SSAR showed that the two values are both 112.8%. Being a slow transient it is expected that these two numbers should be very close as they are in the SSAR. Please explain why they disagree each other in the case of Lungmen?

PSAR Response:

The maximum core average surface heat flux reported in the PSAR in Table 15.0-2 (126.4%) is incorrect and instead refers to the maximum peak surface heat flux. As pointed out in the PSAR Question, for slow transients, it is assumed that the core average surface heat flux and neutron flux are in equilibrium and therefore they should be the same value. Our analysis indicated that the maximum Core Average Surface Heat Flux to be 117.9% of initial. Table 15.0-2 will be revised to show the value of the maximum core average surface heat flux as 117.9% and the value of the maximum neutron flux 117.9% with a note stating, "assuming no scram on APRM thermal power".

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-019

PSAR Sections: 15.0 Accident and Analysis
16.3.3 Instrumentation (Safety System Logic and Control)

Question Date: January 6, 1998

PSAR Question:

In Chapter 15 APRM setdown wasn't described. In page 16.3-8 item 2a, APRM setdown is required. (a) Please clarify whether APRM setdown will be used in the Lungmen project. From the past experience, APRM setdown wasn't an appropriate method. A similar conclusion can also be found on page 2-3 of NEDE-30908P. The statement says, "However, setdown does not provide an absolute guarantee in all cases, and in most cases is operationally over-restrictive." (b) Please clarify whether ARTS (APRM, RBM technical specifications) program will be used in Lungmen project, and why?

PSAR Response:

The APRM Setdown Scram, referred to on page 16.3.3-8 item 2a (Table 16.3.3.1.1-1), is the fixed trip (typically 15% RTP) active with the Reactor Mode Switch in Startup, whose trip functions, will remain in the Lungmen design.

(a) The APRM setdown, for flow biased scram and rod block, was optional as shown in Section, 3.2 Power Distribution Limits, of the BWR 6 Standard Technical Specification. There are currently no APRM setdown requirements in the Lungmen PSAR Chapter 16 which are being evaluated.

(b) The ARTS program currently is not used in the Lungmen design due to the Automatic Thermal Limit Monitor (ATLM) subsystem of the Reactor Controls and Information System (RCIS) providing required protection against violating fuel thermal limits. The Extended Operating Domain program along with other optional features were evaluated for inclusion in the Lungmen NPS.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-020

PSAR Sections: 15.0 Accident and Analysis

Question Date: January 5, 1998

PSAR Question:

1. The safety limit MCPR (SLMCPR) for GE9B fuel is 1.07 in Chinshan while that for the advanced GE12 fuel is 1.09 in the PSAR of Lungmen. Please explain the increase of SLMCPR for the advanced design.
2. What is the reason for the GE12 fuel analysis with TVAPS method to be conservative while that for GE9B fuel in Chinshan is fixed power shape to be conservative?
3. Please provide more description on "uncertainties of power and methodology" in section 15.0.4.4.1. How the statistical adjustment factor was determined for each event? Is it based on uncertainty of power or uncertainty of methodology?
4. Please provide evidence to show that the calculated ΔCPRs are the greatest at EOEC rather than at other exposure points. Is it possible that the minimum IMCPR (initial minimum critical power ratio) is not at the end of cycle, and why?
5. In the analysis of the load rejection with failure of all bypass valves event, the scram signal is turbine control valve fast closure with delay. What is the logic of verification of fast opening bypass valves? What is the required number to verify bypass valve opening as mentioned in section 7.2.2.3?
6. What will be the reactor scram signal for the event of load rejection with one turbine control valve failure in Lungmen? Is it a scram due to control valve fast closure with delay or a scram due to high neutron flux? If the required number to verify bypass valve opening is less than nine, the load rejection delay scram will not be activated. The Lungmen specific ΔCPR for this event can be higher than that in the SSAR. Under such circumstances, how ΔCPR not exceeding 0.18 is assured, and why?
7. Please explain in detail the design philosophy and functions of the combined steam flow limiter, how the upper and lower limits were determined for Lungmen and standard ABWR, respectively, and what are the relationship between these setpoints with full arc and partial arc control.

PSAR Response:

1. GE now calculates the SLMCPR on a cycle specific basis. The number that was provided for the GE12 fuel in Lungmen was based on a preliminary equilibrium core design. As a point of clarification, the SLMCPR for Chinshan is no longer equal to 1.07. For Chinshan 1 cycle 16 it is 1.09, and the number for Chinshan 2 cycle 16 is 1.10.

2. TVAPS is the technically correct procedure for all fuel designs. However, GE performed an evaluation of the effect on fuel designs through GE10 and concluded that there was adequate margin in other parameters, e.g., GEXL correlation, to compensate for any non-conservatism due to the fixed power shape analysis. The US-NRC accepted this explanation and has agreed that TVAPS is necessary only for more advanced designs, i.e., GE11 and later fuel.
3. A statistical adjustment factor, which includes consideration of model bias and uncertainty and an uncertainties for core power. The statistical adjustment factor (SAF) was applied in the ΔCPR results of Lungmen specific events in PSAR Table 15.0-2.

   The ODYN analysis uncertainty for fast pressurization AOOs for Lungmen NPS is given by a statistical adjustment factor (SAF_A) which is defined as:

   $SAF_A = (\Delta CPR/ICPR)_{95/95} - (\Delta CPR/ICPR)_U$

   where

   $(\Delta CPR/ICPR)_U$ = Unadjusted licensing analysis ΔCPR/ICPR based on 100% power and the technical specification scram speed.

   $(\Delta CPR/ICPR)_{95/95}$ = GEMINI Option A 95/95 ΔCPR/ICPR where the "95/95" denotes 95% probability with 95% confidence that the safety limit will not be violated.

   The SAF_A value depends on the transient event and include the uncertainties of power and the ODYN model and varies in the range of 0.003 to 0.01 for Lungmen.
4. For some events, the calculated ΔCPRs are not necessarily the greatest at EOEC. For example, the Loss of Feedwater Heating event was evaluated at BOEC, MOEC and EOEC to determine the limiting exposure. As shown below, MOEC is the limiting exposure in the cycle.

Exposure (MWD/MT)	ΔCPR
0	0.12
4409	0.13
10503	0.10

   For the Mislocated Bundle event, evaluations were also performed throughout the entire cycle and the limiting exposure occurred at 8818 MWD/MT. For the pressurization events, the highest Delta CPR will usually occur at End of Cycle, as all control rods are fully withdrawn thereby minimizing the scram reactivity insertion during the critical stages of the event.

   The minimum initial MCPR is a function of power, flow, exposure, and control rod patterns and the minimum can occur at any point in the cycle.
5. Upon a full load rejection signal, the turbine control valves are fast closed by the turbine control system and the turbine bypass valves are fast opened. When the turbine control valves have fast closed, reactor trip signals are generated based on the inputs from the hydraulic pressure sensors in the supply on the turbine

control valves. A time delay is applied to these signals. During this time delay if the required number of bypass valves are opening as indicated by their 10% position sensors, reactor scram is inhibited.

For the turbine trip and closure of the turbine stop valves, the control actions are similar to that above for full load rejection. Reactor trip signals are generated based on the inputs from the position sensors located on the stem of each turbine stop valve. A time delay is applied to these signals. During this time delay if the required number of bypass valves are opening as indicated by their 10% position sensors, reactor scram is inhibited.

To determine if the required number of bypass valves are opening (for both cases above), the RPS logic needs to know: (1) the reactor power level, and (2) for a given power level, how many bypass valves should be opening. The reactor power level signals with the sufficient accuracy that is required for this purpose, is provided to RPS logic by the NMS (APRM logic). The number of bypass valves that should open for a given power level, will be based on the results of a future analysis that will be performed by GE. The bypass valve interrogation logic will look at the number of 10% bypass valve sensors that have picked up as a function of power level as detected by the APRMs.

6. Upon a full load rejection signal, the turbine control valves (TCVs) are fast closed by the turbine control system and the bypass valves are fast opened. When the turbine control valves have fast closed, reactor trip signals are generated by the hydraulic pressure trip system oil pressure sensors located in the supply to the turbine control valves. As the TCV fast closure RPS scram trip logic is one-out-of-two taken twice, if one TCV failed to close, the RPS signal is still enabled.
7. The basic design of the Maximum Combined Steam Flow Limiter (MCSFL) is the same for Lungmen as it is for the standard ABWR; however, the difference in bypass capacity for Lungmen (110% NBR) versus the standard ABWR (33% NBR) does introduce one difference (described below).

   The MCSFL has two primary functional responsibilities:

   - Lower Limit of the setpoint - The MCSFL must be set high enough such that some bypass flow will be demanded if the pressure control signal increases past the signal which causes the turbine control valves (TCVs) to be wide open. This requirement ensures that enough bypass steam flow can be achieved in the unique event where one TCV is assumed to close and the pressure control signal increases past the value which calls for TCVs to be wide open.
   - Upper Limit of the setpoint - The MCSFL must be set low enough such that the total steam flow produced at this limit is acceptable to the reactor and condenser. The only transient event which challenges this limit is the unlikely upscale failure of the triplicated pressure control signal such that it tries to open all the turbine and bypass valves (BPVs). There is no significant challenge to the reactor fuel in this event, but mitigation is required to ensure that the potential depressurization and cooldown of the

reactor is acceptably limited and that adequate control of pressure and water level is recovered. Another concern from an availability standpoint is that a high MCSFL could cause such a high steam flow condition to exist in the main steam lines, that the MSIVs would auto isolate, thus isolating the reactor.

In this pressure regulator failure situation described above, there is a difference between the Lungmen full bypass unit and the standard partial bypass ABWR. With partial bypass capacity, it can be shown that this failure can be allowed to fully open the TCVs and the BPVs and still have acceptable consequences with respect to stopping the depressurization and cooldown, and recovery of the unit to normal water level and pressure conditions. In this situation, there is really no Upper Limit requirement on the setpoint, and the setpoint calculation and surveillance requirements are simpler. The standard ABWR design has chosen to take this simpler approach.

For Lungmen, full opening of all of the TCVs and BPVs is very unlikely because of the triplication of the control design and because the normal spanning of the pressure control signal does not enable the system to produce such a signal. However, to provide additional assurance that such a large depressurization disturbance is not created, the current Lungmen design has established an Upper Limit of 130% steam flow for the setpoint. This means that the TCVs would possibly be fully opened (~105% steam flow) and the signal producing BPV opening would be restricted to only 25% BPV flow. Transient analysis for all BWRs has shown that such a limiter setting will constrain the consequences of the event such that it will remain within the thermal duty design of the reactor and that acceptable pressure and water level control can be reestablished during the event.

Concerning the relationship between the maximum combined flow limiter setting for full and partial arc control, this was addressed only in the SSAR. The standard ABWR documentation has indeed discussed two values for the Lower Limit of the MCSFL - 115% and 125%. These values came from two different assessments of the transient event in which one TCV is assumed to close.

One assessment assumed a relatively large, full arc turbine in which the capacity of the three remaining TCVs was approximately 95% steam flow. In this situation, conservative analysis showed that acceptable results were calculated if the MCSFL was 115% (or higher). This setting would achieve about 12% BPV flow in addition to the 95% passing through the three open TCVs.

The other assessment assumed a more standard, partial arc turbine in which the capacity of the three remaining TCVs was approximately 85% steam flow. In this situation, conservative analysis showed that acceptable results were calculated if the MCSFL was 125% (or higher). This setting would achieve about 22% BPV flow in addition to the 85% passing through the three open TCVs. Thus the overall results for the two cases were about the same.

The dominant factor for selecting the Lower Limit for the MCSFL is the steam flow capability of the three remaining valves, not whether the plant is designed with partial arc or full arc controls. The Lungmen turbine design is expected to be more like a standard turbine with about 105% flow with four TCVs wide open and about 85% flow with only three TCVs fully open.

The first assessment for Lungmen considered setting the MCSFL at 125% as implied by the previous analyses. However, Lungmen-specific analysis of the one TCV closure event has shown that acceptable results are obtained with the MCSFL set as low as 115%. Actually a sensitivity analysis study was performed to find the worst case ΔCPR for the 115% through 130% maximum combined flow limiter setting. As reported in the PSAR the 115% setting yielded a ΔCPR of 0.15 and the 130% setting yielded a ΔCPR of 0.07. This analysis will be repeated as the final plant design is established, but for now this setting limit has been selected from the Lungmen analysis. It provides a better range for setpoint adjustment (which is more difficult when a setpoint has two-sided requirements).

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

ROCAEC Review Comments:

This question comprises of 7 independent items and items 2 and 5 are not accepted for the moment.

2. The response stated that "GE performed an evaluation of the effect on fuel design through GE 10 and concluded that there was adequate margin in other parameters". Please provide the evaluation report or document.

   Items 3 and 4 are accepted. It is suggested that the relevant data provided in the response be used extensively in FSAR.

5. Please clarify the following :

   1) Is the signal that verifies the bypass valve is 10% open done by microswitch ? How many signals are there ? Has single failure been considered ? Is it possible that bypass capacity becomes insufficient when bypass valve opens more than 10% but stopped before reaching 100% ?

   2) Number of bypass valves to be verified to be open to avoid scram will be different under different power ratings. Please use examples to illustrate the operations. e.g., under 80% power, if turbine tripped, 10 bypass valves are required to be open or only several bypass valves are required to be open ?

3) In the future FSAR, is the relevant analysis to verify bypass valve openness going to be performed under 100% power only or under different power levels ?

Further Clarification:

2. See Attachment C of NEDE-32417P, GE12 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II), December 1994 for the discussion of TVAPS. This reference was provided earlier.

5. Concerning the bypass valves,
   1) The signal that verifies that a bypass valve is open is done by valve limit switches. Each bypass valve has 4 limit switches and the logic is 2 out of 4. The bypass valve interrogation logic looks at the 10% position as indicating that the valve is indeed opening and it expects that the valve will fully open based on the bypass valve quick opening characteristics. Bypass valve capacity could become insufficient if the valve opens to 10% but fails subsequently to reach 100% open.
   2) It is expected that the logic will use a power dependent algorithm with load rejects / turbine trips at less than 100% rated to open proportionally and simultaneously the required number of bypass valves. For example a 80% power turbine trip will open approximately 8 bypass valves.
   3) It is expected that some sensitivity study will be performed at various power levels.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-021

PSAR Sections: 15.6

Question Date: March 9, 1998

PSAR Question:

1. The MSIV leakage (total all lines) specified in SSAR was 6.1 L/min but in Lungmen PSAR it is 21.7 L/min. Please explain the difference between these two.

2. Regarding Table 15.6-8 :
   (1) Please explain what values should the plateout and resuspension factors be under the item MSIV leakage ? and how are they determined ?
   (2) Please explain how the Iodine Removal factor under the condenser data item is determined.
   (3) Please explain what the advantages and disadvantages are for the Lungmen design dealing with the MSIV leakage question compared with the leakage control system (LCS) used in other BWRs.

3. Please clarify what the objects are in Figure 15.6-2 (see attached figure with arrows indicated).

Response:

1. The value in the SSAR is a typographical error and should read 66.1 L/m which is based upon a value of 35 standard cubic feet per hour per line or 140 scfh total. The Lungmen NPS MSIV leakage was reduced to 21.7 L/m total or 11.5 scfh per line for a total of 46 scfh to provide further mitigation of MSIV leakage due to the proximity of the Lungmen NPS power block buildings to the site boundary. The value of 11.5 is a standard for many BWRs built before ABWR.

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2. The following is GE proprietary information.

(1) The equations used to determine steam line and drain line inorganic iodine removal and resuspension of iodine plated out onto these surfaces are:

Inorganic iodine is plateout onto steam and drain lines according the equation based upon work done by Genco (ref. 2-1).

$$V_d = 9x10^{-8}\ e^{(8100/RT)}$$

where:

V_d is the deposition velocity in cm/s

R = 1.987

T is the pipe surface temperature in K.

Material plated on a steam or drain line may be permanently fixed to the surface by the equation based upon unpublished NRC research (see PSAR Reference 15.6-4) :

$$\text{fixation}(T) = 1.351\text{x}10^{-4}\, e^{(-1185/T)}$$

where fixation is a rate constant specifying the rate at which iodine plated out onto a surface becomes fixed to that surface.

In addition iodine plated out onto a pipe may also undergo a chemical reaction converting the inorganic iodine to a organic form resulting in resuspending the iodine as an organic species according to the equation:

$$\text{respen}(T) = 1.246\text{x}10^{-6}\, e^{(-600/T)}$$

where respen is a rate constant specifying the rate at which the iodine is converted and removed from the surface.

The above values are based upon both empirical and analytical studies as outline in the referenced document and were subject to NRC review and approval.

(2) The derivation of the terms for inorganic iodine removal in a condenser are given in section 7.1.5 of PSAR Reference 15.6-4 and results in the equation:

symbol 108 \f "Symbol" \s 11$\lambda_n = K_g A/V$

where:

symbol 108 \f "Symbol" \s 11λ_n is the removal rate constant due to surface deposition,

K_g is the average mass transfer coefficient,

A is the surface area for wall deposition, and

V is the volume of contained gas.

K_g is defined as:

$K_g = 0.13\, D\, (G_r\text{-}S_c)^{1/3} / L$

where:

D is the diffusivity of iodine in gas phase,

G_r is the Grashov number,

S_c is the Schmidt number, and

L = Length measured along the deposition surface.

End of Proprietary Information

(3) The introduction of Leakage Control Systems in the late 1970's was in response to NRC Safety Issue C-8 (please refer to PSAR subsection 1C.2.61.1 for further discussion on this issue). Shortly after the installation of these systems into most BWR's, the BWR Owner's Group identified the LCS as both a significant maintenance cost item as well as a candidate for ALARA consideration. In addition, during that time period MSIVs were found with larger than acceptable leakage rates which would exceed the design of the LCS, rendering the LCS ineffective. The BWROG MSIV program committee looked at eliminating the LCS by replacing the LCS with a passive, low maintenance, low ALARA system. The result was a generic procedure for insuring the adequacy of the passive leakage components using analysis and seismic expert walk downs, and a methodology to conservatively evaluate potential control room and offsite doses. Since then eight BWR's in the U.S. have removed their LCS in favor of the passive mitigation methods referred to here in addition to licensing ABWR in the U.S. using the same methodology

3. Figure 15.6-2 is a generic pathway schematic of potential leakage in the ABWR. The pathway indicated in the lower portions of the ABWR secondary containment represent the release of contaminated fluids from safety related equipment such as RHR pumps and evolution of the fission products into the atmosphere of the plant and finally to the SGTS system. Such a pathway is considered in the Lungmen analysis and is found to be negligible when compared to the unspecified leakage pathway from the containment and when compared to the MSIV leakage pathway.

Please refer to track number 15-022 for further clarification.

References:

2-1. BMI 1863, Fission Product Enhancement Under Reactor Accident Conditions: Deposition on Primary Surfaces, J.M. Genco, W.E. Berry, H.S. Rosenberg, and D. Morrison, Battle Memorial Institute, March 1969.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-022

PSAR Sections: 15.6.5.5

Question Date: March 9, 1998

PSAR Question:

Activity/release was provided in Table 15.6-9, 15.6-10, 15.6-11, 15.6-12. It was found that there is no description associated with these tables. Please provide adequate descriptions about these tables, including but not limited to the following assumptions, analytical tools, methodology, reference and compliance with regulatory requirements.

Response:

Table 15.6-10 and 15.6-12 are referenced in paragraph 15.6.5.5.3.2, but there is no description of these tables or of Tables 15.6-9 and 15.6-11 since it was felt that the contents were self evident and described in detail in References 15.6-2 and 15.6-4. The following is a summary of the contents of these tables and is applicable to all the radiological tables of similar nature appearing in chapter 15.

Table 15.6-9 provides inventories of those radiologically significant iodine isotopes used in the analysis for regulatory compliance. Quantities of these isotopes are released to the containment at time t=0 in accordance with the data in Table 15.6-8 and these tables then show the inventories of those isotopes in each area of the pathway analysis (see Figure 15.6-2) valid for the time indicated. Notice that the tables also break down the species of the iodines into (1) elemental and particulate (which are treated alike) and (2) organic. A simplified homogenous mixing model utilizing radioactive decay is used to model the separate compartments (containment, secondary containment, condenser, Turbine Building, and Main Control Room). The equations and details of this modeling are provided in Reference 15.6-2 for the primary containment and secondary containment. Included in this table are the results of an analysis for steam line leakage in accordance with Reference 15.6-4 which is currently used to determine steam line leakage consequences for design basis accidents (see response to Question 15-021). This model transports the various species of iodines down the steam and drain lines into the condenser and to the Turbine Building and then offsite or to the intake to the Main Control Room. The model provides radioactive decay for the organic iodines in the steam and drain lines and mixing and decay in the condenser. The non-organic iodines are deposited in transport down the steam and drain lines resulting in the fixation[1] of the material onto the lines or a chemical reaction on the

[1] Basically two inventories are kept internally by the model, (1) the amount deposited on the pipe and (2) the amount which has become permanently fixed to the pipe and is removed from

lines and subsequent re-evolution of the iodine as an organic species for transport through the system to the environment (again see the response to Question 15-021). Therefore the section of this table for the condenser also adds the inventory of resuspended iodines. Please note in this table that for the first two time periods no activity is shown in the condenser for the non-organic and organic iodines. In fact the model predicts that these species are in transit in the steam lines and do not begin to appear in the condenser until shortly before one hour has elapsed. The resuspended iodines do show almost instantly because the model assumes that these isotopes are instantly transported to the condenser (unlike the other two species) as a conservatism and to simplify modeling. Complete details of this modeling are given in Reference 15.6-4. Finally, section D of the table shows the inventory at the end of the time period for each isotope followed by a second table which shows the integrated activity in the Main Control Room for that time period. Since the concentration of each isotope varies over a given period, it is the integrated activity (in MBq-s) which is used to provide the dose calculation.

Table 15.6-11 is the sister table to 15.6-9 in that it provides equivalent information on the radiologically significant noble gas isotopes. For the steam line modeling, noble gases can only be delayed in transit and therefore deposition and resuspension models for steam and drain line transit are irrelevant and not included.

Tables 15-6-10 and 15.6-12 provide integrated releases to the environment of the separate isotopic species shown in Tables 15.6-9 and 15.6-11. Note that the Main Control Room integrated activities given in Tables 15.6-9 and 15.6-11 are provided for the individual time periods whereas the integrated releases provided in Tables 15-6-10 and 15.6-12 are an integration over total time up to the point shown. This is done in the case of Tables 15-6-10 and 15.6-12 to show the increase in time of the fission product release to the environment. Individual activity integration is submitted in Tables 15.6-9 and 15.6-11 to provide detail on risk and dose rate which can be derived from these values for operator exposure.

Exact details on modeling and assumptions are provided in the two referenced documents. As to compliance with regulatory requirements, Reference 15.6-2 represents GE's interpretation of models and analysis methods stipulated by regulatory requirements and has been used by GE since 1972 for SAR submittals and analysis. This document (and its revisions) and similar documents are not reviewed and approved by the NRC since the NRC does not accept external models and documentation in the area of radiological compliance. Rather the NRC compares SAR analysis to NRC internal calculations to reach conclusions of regulatory compliance. As to Reference 15.6-4, this document and the associated radiological modeling have received very extensive NRC review prior to acceptance.

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further assessment. Fixation refers to the chemical process by which the deposited material becomes permanently fixed to the pipe (see the response to Question 15-021).

RESPONSES TO ROC-AEC's PSAR QUESTIONS

ROCAEC Review Comments:

Not accepted for the moment.

1. Please explain in PSAR why the iodine activity in the condenser is a lot smaller than what is in the SSAR. Even though the leakage rate of MSIV in PSAR is only 1/3 of SSAR but it still does not explain the difference. Also, response to Question 470.4 of chapter 20 of SSAR (Table 20.3.1-1) showed that the effects of Condenser plateout and hold-up are very significant. What has been considered in PSAR ? plateout and/or hold-up ?

2. Table 20.3.1-1 of SSAR showed that the thyroid dose at LPZ will be very high if iodine is treated according to RG 1.3 but PSAR showed a value lower than what in that Table. Please clarify. (Note : PSAR showed a LPZ value of 300 m which is shorter than the 800 m in Table 20.3.1-1)

Further Clarifications:

1. The processes by which the transport to the condenser of iodine isotopes is highly non-linear being a combination of transport velocity which is proportional to leakage rate combined with steam line and drain line temperature which are an exponential function of material properties combined with initial temperature and insulation thickness of the lines. These provide input to the three exponential equations provided previously which determine deposition velocities in the piping along with fixation rates and resuspension rates of the iodine as it is transported down the piping to the condenser. Therefore a one on one comparison with the ABWR SSAR would not be useful. As to the response to Question 470.4, the MSIV model used considers both plateout and hold up. For elemental and particulate iodine, plateout is the primary mode of removal in the piping lines with hold up a minor contributor. For the organic iodine, hold up is the only mitigation mode.

2. Table 20.3.1-1 should not be used for any comparisons to the values shown in Chapter 15 of the SSAR or the Lungmen PSAR. Table 20.3.1-1 was developed for the NRC to provide a sensitivity study for the initial submittal of the SSAR made in 1989 which was not a Regulatory Guide 1.3 compliant evaluation. After Table 20.3.1-1 was submitted, the evaluations found in Chapter 15 were replaced with Regulatory Guide 1.3 compliant evaluations at the direction of the NRC. Therefore there is no basis in comparisons between those evaluations found in Chapter 15 and Table 20.3.1-1. A version of Table 20.3.1-1 for use on the Lungmen PSAR submittal is in the process of being prepared and will be submitted by August 6.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

INER Question at July 10, 1998 Meeting with the ROCAEC:

Why the inventory tables are different between the ABWR SSAR and the Lungmen PSAR?

Clarification:

There are two reasons why the inventories in tables 15.6-9 through 15.6-12 would be different between the ABWR SSAR and the Lungmen PSAR. The first reason which is applicable to short time periods is demonstrated in the following table. In this table values for I-131 and Xe-133 are converted from the original ABWR SSAR computer code run, first from a three digit precision to a two digit precision which was printed in the SSAR Amendment 31. Then the values (with two digit accuracy) were converted from curies (in amendment 31) to megabecquerels in amendment 34. When the results were calculated in curies, converted to MBq, and then rounded, the values shown under "Direct conversion and round off" resulted. The Lungmen PSAR results with the same method shows the same values at the early time (1 min).

Original Computer Code Output ABWR SSAR

Curies	1 min	4 days	30 days
I-131	1.37E+07	9.24E+06	7.16E+05
Xe-133	2.21E+08	1.25E+08	2.97E+06

Amendment 31 ABWR SSAR

Curies	1 min	4 days	30 days
I-131	1.4E+07	9.2E+06	7.2E+05
Xe-133	2.2E+08	1.2E+08	3.0E+06

Amendment 34 ABWR SSAR

MBq	1 min	4 days	30 days
I-131	5.2E+11	3.4E+11	2.7E+10
Xe-133	8.1E+12	4.4E+12	1.1E+11

Direct conversion and round off ABWR SSAR

MBq	1 min	4 days	30 days
I-131	5.1E+11	3.4E+11	2.6E+10
Xe-133	8.2E+12	4.6E+12	1.1E+11

Direct conversion and round off Lungmen PSAR

MBq	1 min	4 days	30 days
I-131	5.1E+11	3.5E+11	3.1E+10
Xe-133	8.2E+12	4.7E+12	1.3E+11

The second reason that the ABWR SSAR and the Lungmen PSAR do not show the same long term inventories is that the analysis conditions are different. All the calculations begin with the primary containment inventory which is initially the same for ABWR and Lungmen. But as time progresses, the Lungmen inventory for the primary containment should not fall as rapidly as ABWR since the MSIV leakage for Lungmen (46 scfh) is less than the ABWR (140 scfh) so the inventories should diverge over time.

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

ROCAEC Review Comments:

(1) Please further explain why the Lungmen MSIV Leakage Rate is 1/3 of SSAR but the Iodine activity at condenser after one hour is only about 1/100 of that of SSAR value? Any obvious differences between the calculation models of the two?

(2) Please clarify how the iodine spike is determined in PSAR Table 15.6-1? Does it satisfy the requirement of 500 times of the equilibrium values as in SRP?

(3) Please explain how the initial core inventories are calculated.

(4) Please provide information of the decay scheme used in calculating the isotopes activity including I-131, I-132, I-133, I-134, I-135, Kr-83m, Kr-85m, Kr-85, Kr-87, Kr-88, Kr-89, Xe-131m, Xe-133m, Xe-133, Xe-135m, Xe-137 and Xe-138.

(5) Please provide the initial core inventories and half-life of the parents for the above isotope dose calculation.

Further Clarifications:

(1) There is no difference in the models used between the final issue of the ABWR SSAR and in the Lungmen PSAR. The only difference is in the values for the parameters used. Specifically, the comment refers to the difference in leakage rates being approximately a factor of three. In fact the analyses are based on higher difference. For the ABWR SSAR the MSIV leakage rate in SSAR Table 15.6-8 is listed at 66.1 L/min (140 scfh) and in the Lungmen PSAR the same table lists 21.7 L/min (46 scfh). But when GE made the SSAR calculations, the value used in the analysis was based upon an extremely conservative assessment of the fractional leakage from the containment. Specifically, GE codes input leakage in terms of per cent of inventory per day leakage. For the SSAR the value used was 0.719 % per day based on dividing the 66.1 L/min by the volume of the containment. By calculating the leakage in this fashion, the results provided are more conservative and permit some latitude in design modifications which is necessary when working with a generic design. For the Lungmen PSAR, the calculated value was 0.12% per day which is roughly a factor of six smaller than the ABWR SSAR. The Lungmen calculation follows the method recommended for detailed plant analyses and includes consideration of the containment design pressure and temperature since the leakage is specified in terms of standard

volumetric units. What is meant is that the 21.7 L/m is the volumetric leakage rate at 20 symbol 176 \f "Symbol" \s 11阿C and one atmosphere, therefore the actual leakage rate must be back calculated to the design temperature and pressure of the containment. Therefore the only factor different between the ABWR SSAR and the Lungmen PSAR in this calculation was the activity leakage rate from the containment which was a factor of six different.

To understand what this difference in leakage rates means, the following example is submitted. This example looks at the doses for the time period from 96 to 720 hours for two reasons:

1) A rough equilibrium is reached by that point in the accident. Before then, steam and drain line temperatures are changing rapidly along with flow rates caused by the steam line temperatures and isotopic inventories.

2) The majority of the thyroid dose is delivered in that interval.

The following table provides a dose comparison between the ABWR SSAR and the Lungmen PSAR. Values from the SSAR and PSAR are compared by time period and by removing the meteorology values. This is done by the following method:

In the following table the values found in the last four rows are calculated by:

Example using the Reactor Building Pathway:
Whole body value at 8 hours, 0.99

1. Determine the incremental dose for the SSAR and Lungmen
   SSAR = 0.0102 - 0.00464 = 0.00556
   Lungmen = 0.00517 - 0.00355 = 0.00162
2. Divide each dose by the respective meteorology
   SSAR = 0.00556 / 1.56E-4 = 35.64
   Lungmen = 0.00162 / 4.62E-05 = 35.06
3. Ratio Lungmen by SSAR = 35.06 /35.64 = 0.984

Comparison of LOCA Dose Results Between ABWR SSAR and Lungmen PSAR

ABWR SSAR Certification

Total	2	8	24	96	720
Whole Bod	4.66E-03	1.07E-02	1.80E-02	2.95E-02	3.71E-02
Thyroi	2.15E-01	3.04E-01	5.09E-01	1.28E+00	2.42E+00
MSIV Pathway					
Whole Bod	1.76E-05	4.80E-04	1.59E-03	4.80E-03	8.59E-03
Thyroi	2.65E-04	1.78E-02	8.39E-02	4.81E-01	1.37E+00
Reactor Building Pathway					
Whole Body	4.64E-03	1.02E-02	1.64E-02	2.47E-02	2.85E-02
Thyroid	2.15E-01	2.86E-01	4.25E-01	8.00E-01	1.04E+00
Meteorology	*1.56E-04*	*1.56E-04*	*9.61E-05*	*3.36E-05*	*7.42E-06*

Lungmen PSAR

Total	2	8	24	96	720
Whole Body	3.55E-03	5.18E-03	7.12E-03	1.06E-02	1.72E-02
Thyroid	1.66E-01	1.87E-01	2.32E-01	4.60E-01	1.96E+00
MSIV Pathway					
Whole Body	2.18E-07	1.07E-05	7.69E-05	9.01E-04	6.27E-03
Thyroid	3.57E-06	4.02E-04	4.34E-03	1.12E-01	1.53E+00
Reactor Building Pathway					
Whole Body	3.55E-03	5.17E-03	7.05E-03	9.68E-03	1.09E-02
Thyroid	1.66E-01	1.87E-01	2.27E-01	3.48E-01	4.32E-01
Meteorolog					
ground	*3.95E-04*	*3.06E-04*	*2.69E-04*	*2.04E-04*	*1.37E-04*
elevated	*1.19E-04*	*4.62E-05*	*2.88E-05*	*1.04E-05*	*2.38E-06*

Comparison of ABWR SSAR to Lungmen PSAR LOCA Doses
Ratio of Doses Lungmen/SSAR for incremental time period.

MSIV Pathway					
Whole Body	0.0049	0.0115	0.0213	0.0423	0.0768
Thyroid	0.0053	0.0116	0.0213	0.0447	0.0862
Reactor Building Pathway					
Whole Bod	1.00	0.99	1.01	1.02	1.04
Thyroi	1.01	1.00	0.98	1.04	1.08

In addition to the difference in MSIV activity leakage rates, the condenser leakage rate will also change since the driving force in the condenser is also the MSIV leakage. ABWR SSAR condenser leakage stands at 11.6% per day (AWBR SSAR Table 15.6-8, II e)and Lungmen at 3.6% per day, (Lungmen PSAR Table 15.6-8, II e) which is the rough factor of three since we are now comparing standard units of volume.

To evaluate the difference caused by the change in leakage rate, a good choice is a fission product such as Kr-85 with a 10 year half life. Kr-85 does not plate out or chemically bond in transport down the steam and drain lines and is subject only to hold up and decay (of which there is little due to the long half life) so that in fact the MSIV pathway is a two compartment model (containment and condenser) with an intermediate decay (about an hour in going down the steam and drain lines). The Kr-85 released from the condenser in the period 96 to 720 hours in the ABWR SSAR is 5.67E9 MBq and in Lungmen is 5.38E8 MBq. Using a simple two compartment equation (shown below) and the transport values given above and ignoring radioactive decay, one finds that the Lungmen values should give a release about 0.091 of that of SSAR which is very close to the Kr-85 releases. This means that for long periods of time, just due to the lower leakage rate, Lungmen values should be 10% of ABWR SSAR without considering radioactive decay.

Simple 2-Compartment Release Equation

$$\mathrm{Re}\,lease = \frac{L_{cndsr} L_{msiv}}{L_{cndsr} - L_{rb} - L_{msiv}} \left(\frac{1 - e^{-(\lambda + L_{rb} + L_{msiv})t}}{\lambda + L_{rb} + L_{msiv}} - \frac{1 - e^{-(\lambda + L_{cndsr})t}}{\lambda + L_{cndsr}} \right)$$

Lx is leakage rate for:
cndsr - condenser
msiv - main steam line valve leakage
rb - leakage from primary to secondary containment

On a further note, it is interesting to note that when radioactive decay is taken into account using a two compartment model, one finds that the values are reasonable. Using a I-131 half life, the two compartment model shows a difference of 0.0834 which tracks closely with the 0.0862 ratio comparing the 96-720 hour SSAR-Lungmen ratio above. In other words, the dominating factor in the change from SSAR to Lungmen can be explained solely by the change in MSIV leakage rate.

For time periods shorter than the 96 hours a direct comparison is significantly more difficult. This owes to the fact that the calculation varies significantly as the steam and drain lines cool. In comparison to SSAR the flow rate into the steam lines for Lungmen is one sixth the ABWR SSAR rate, consequently, the flow velocity which varies as the steam line temperature change is also significantly different. With lower flows and lower flow rates, the plateout will increase but not necessarily the resuspension since resuspension is a function of temperature and even though the total mass plated out is increased, the resuspension may also be suppressed because of the lower temperatures. As an example, flow into the steamlines at the initiation of the accident for Lungmen will take 1.12 hours before being observed at the condenser. For the ABWR SSAR, this same release requires 0.6 hours, therefore the fission products in SSAR will undergo less plateout, but those products which are plated out will see higher temperatures and the resuspension rates will be higher. At twenty-four hours SSAR flow will take 40 minutes to transit the lines and Lungmen will take 77 minutes. The overall difference cannot be summarized by short studies as above.

Nevertheless, the dominant contribution to dose via the MSIV is in the period beyond 96 hours and a simple analysis as performed above shows that the difference in leakage rate between SSAR and Lungmen adequately explain the overall differences.
MSIV leakage is tested in accordance with the specification of PSAR subsection 5.4.5.4.

(2) GE believes that the 500 times factor for iodine spiking found in SRP 15.6-2 is a gross over simplification of the spiking process and without any merit or sound scientific backing. GE is not using now nor has ever used the factor. Rather, GE has developed a spiking process based upon a study of GE fuel and dependent upon the depressurization process as the plant is shut down. This method of analysis has been submitted by GE in all their past FSARs and found to be an acceptable alternative by the NRC. The model is described in GE document NEDO-32708, Chapter 7 which has been supplied.

The spiking values found in the analysis were derived by looking at 7 x 7 fuel in the early 70 and calculating spiking releases from this fuel type. A distribution was obtained and values chosen such that the values used for the spiking releases bounded 95% of the experience in this fuel. The values used then present a bounding on current fuels since based upon the understanding that current fuel performance in areas such as fuel leakers significantly exceed the experience in 7 x 7 fuels then the assumed 7 x 7 fuel spiking release would also bound current fuel spiking releases.

As the NRC has reviewed this spiking model used by GE in past FSARs as well as in the ABWR SSAR and accepted the results, it does satisfy the requirements of SRP 15.6-2.

GE will provide in the FSAR a comparison using GE-12 fuel to prove that the spiking releases used by GE are applicable to GE-12 fuel.

(3) The mathematical model used to derive the core inventory is explained in Los Alamos National Laboratory report LA-5885-MS, which will be provided. See item 5 for further information.

(4) We assume that by decay scheme what is meant is the parent daughter relationship used in calculating the basic core inventory and not any parent daughter relationships in calculating inventories in separate areas post-accident. GE does not employ parent daughter schemes for post-accident analysis. For decay schemes for core inventory calculations, the data is currently in the form of a 14,000 line Cinder data input deck. The deck (or the information derived from the deck by hand) can be supplied, if it would be useful to the ROCAEC. Four weeks would be required following the request.

(5) The inventories were provided in response to Comment 15-035 and are repeated here. Inventories are provided for isotopes of Krypton, Xenon, and Iodine for a power level of 3926MWt and are based upon the GE Standard BWR Inventory. This inventory was calculated using a GE version of the CINDER code from Los Alamos National

Laboratory using an ENDF B/IV data base assuming a core average burn up of 29,200 MWD/t. The inventory was calculated for a 1000 day burn and is conservative for Lungmen based upon the standard radiological isotopes indicated by an "x" in the table below

Lungmen NPS Core Inventory at Shutdown

Isotope	Stnd	Decay Constant (/sec)	MBq
KR-81m		5.21164E-02	1.734E+05
KR-83m	x	1.03517E-04	4.557E+11
KR-85	x	2.04702E-09	4.380E+10
KR-85m	x	4.29779E-05	9.782E+11
KR-87	x	1.52006E-04	1.876E+12
KR-88	x	6.87647E-05	2.658E+12
KR-89	x	3.65584E-03	3.306E+12
KR-90	x	2.14597E-02	3.326E+12
KR-91		7.96722E-02	2.497E+12
KR-92		3.76711E-01	1.268E+12
XE-129m		1.00282E-06	1.526E+04
XE-131m	x	6.69100E-07	2.298E+10
XE-133	x	1.51655E-06	8.030E+12
XE-133m	x	3.59755E-06	3.348E+11
XE-134m		2.39016E+00	5.749E+10
XE-135	x	2.09968E-05	1.038E+12
XE-135m	x	7.55063E-04	1.514E+12
XE-137	x	3.00845E-03	7.048E+12
XE-138	x	8.13554E-04	6.696E+12
XE-139	x	1.71571E-02	5.265E+12
XE-140		5.09668E-02	3.492E+12
XE-141		4.02993E-01	1.273E+12
I-128		4.62099E-04	5.676E+07
I-129		1.38237E-15	1.266E+05
I-130		1.55275E-05	1.519E+11
I-130m		1.29803E-03	1.112E+11
I-131	x	9.97705E-07	3.821E+12
I-132	x	8.42630E-05	5.586E+12
I-133	x	9.25678E-06	7.992E+12
I-133m		7.70164E-02	2.304E+11
I-134	x	2.19629E-04	8.798E+12
I-134m		3.20902E-03	8.146E+11
I-135	x	2.92393E-05	7.546E+12
I-136		8.35118E-03	3.638E+12
I-136m		1.44406E-02	2.078E+12
I-137		2.81767E-02	3.613E+12
I-138		1.06638E-01	1.844E+12
I-139		2.88812E-01	8.539E+11
I-140		8.05986E-01	2.611E+11

Isotopes with a "x" in column "stnd" are used

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **15-023**

PSAR Sections: **15.6.5.5.1**

Question Date: **March 24, 1998**

PSAR Question:

In SRP section 15.6.5, Appendix A, it is required to include the following sources and radioactivity transport paths:

contribution from containment leakage
contribution through containment purge/vent valves during closure
contribution from post-LOCA leakage from ESF system outside containment
contribution from main steam isolation valve leakage

We didn't find the description about items (2) and (3) listed above. Please provide the description and discuss its impact on the results of radioactivity analysis after LOCA.

In section 15.6.5.5.1 of PSAR, the following statement is given: "All leakage pathways from the primary containment, except the main steamlines and the feedwater lines terminate in the Reactor Building." Please clarify whether the pathway from the drywell sump to the Radwaste Building be considered as a leakage pathway or not.

Response:

With respect to containment leakage:

(1) leakage via purge and vent valves is treated as unspecified containment leakage as these valves are directly connected to the SGT system or are isolated during an accident by secondary containment isolation valves.

(2) leakage from ESF systems post-LOCA is a minor contributor to the total leakage and was treated as negligible. This was also a question from the NRC during ABWR certification and a detailed assessment was provided in response to that question. That response and the reply to the purge/vent valve question will be added to the PSAR to provide full compliance with SRP 15.6-5, Appendix A.

The statement referred to in section 15.6.5.5.1 is incorrect in that the drywell sump discharge lines are routed directly to the Radwaste Building. The statement will be amended to indicate this pathway and a limiting analysis provided in the PSAR.

ROCAEC Review Comments:

Please revise the response to incorporate the new statement to be amended in PSAR and the analysis committed in 15-023 original responses.

Further Clarifications:

With respect to containment leakage as discussed in 15.6.5.5.1, GE has reviewed the design of the primary containment sumps in the lower drywell. These sumps are of two types: (1) for specified leakages from specific equipment and (2) a general drain sump for unspecified leakages. Both sumps penetrate the primary containment boundary and are isolated by two motor operated isolation valves each, one inside containment and one outside containment. Both lines are interconnected with multiple sump drain lines in the reactor building prior to exiting the reactor building into the radwaste tunnel which is uphill from the lower reactor building tunnel exit by several meters of vertical height: GE assessment of these lines is that the drywell sumps will be isolated from the secondary containment in accordance with standard practice and that the lines leading from these isolation valves will most certainly be water filled and at pressure due to the delta in pressure from the vertical lift in the lines. Therefore, any potential leakage from these lines is considered to be minimized by the design and negligible when compared to the primary pathways for leakage.

The statement referred to in section 15.6.5.5.1 will be amended with an explanation of negligible potential leakage from the sump lines as follows:

> *"All leakage pathways from the primary containment, except the main steamlines, the feedwater lines and the drywell sump discharge lines, terminate in the Reactor Building. Any potential leakage from the drywell sump discharge lines is considered to be minimum by their design and negligible when compared to the primary pathways for leakage. There are two sump lines: (1) for specified leakages from specific equipment and (2) a general drain sump for unspecified leakages. Both sump lines penetrate the primary containment boundary and are isolated by two motor operated isolation valves each, one inside containment and one outside containment. Both lines are interconnected with multiple sump drain lines in the reactor building prior to exiting the reactor building into the radwaste tunnel which is uphill from the lower reactor building tunnel exit by several meters of vertical height: The drywell sump lines will be isolated from the secondary containment in accordance with standard practice and the lines leading from these isolation valves will be water filled and at pressure due to the difference in pressure from the vertical lift in the lines.* Leakage through the steamlines is treated..."

As mentioned in Item (2) of the initial response, a section will be added to the PSAR as follows:

15.6.5.5.1.4 ECCS Leakage

Leakage from engineered safety features are not specifically analyzed. The total leakage from the primary containment is restricted to 0.5% per day for all leakage except that through the main steam line isolation valves. Leakage from engineered safety features is then included in the 0.5% per day such that all leakage from equipment external to the primary containment shall not result in an airborne release which when combined with the containment leakage shall result in an equivalent release greater than 0.5% per day.

The following is a first order estimation of the effect of ECCS Leakage on Total Primary Containment Leakage.

(1) Leakage was estimated from the ABWR ECCS systems including the RHR and HPCF. The RCIC was ignored as a minor contribution since the RCIC will cease to operate on decrease of steam flow prior to any fission product release. The leakages from the systems were:

(a) 35 RHR valves
10 HPCF valves
45 valves total, total valve steam leakage = 7.57E–08 m^3 /s

(b) 3 RHR pumps
2 HPCF pumps
5 pumps total, total normal pump leakage = 2.78E–08 m^3 /s

(c) Total ECCS leakage = 1.03E–07 m^3 /s

(2) Primary Containment Leakage

0.5%/day * 25% core inventory = 0.00125 core fraction/day

(3) ECCS Leakage

0.5 core inventory/3,899 m 3 water = 1.28E–04 core frac/m^3

1.28E–04 * 1.03E–07 * 60 * 60 * 24 = 0.00000114 core frac/day

flashing = 0.4

Total airborne release from ECCS = 0.00000114 * 0.4 = 0.000000456 frac/day

It is concluded that the ECCS leakage is a small fraction of total primary containment leakage

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **15-024**

PSAR Sections: **15.7.1.2**

Question Date: **March 10, 1998**

PSAR Question:

NUREG-0016 uses British units so equation 1.5.1.6 becomes
$T = 0.262 * Mk_d / (10N)$
but the Adsorption equation (11.3-1) in Section 11.3.3.2.4 is
$T = Mk_d / V$
Are they consistent ?

Response:

Yes, the equation for the determination of the value of T given in 1.5.1.6 is described in paragraph 2.2.9.2 of NUREG-0016. In looking at the derivation of this equation, the assumption that the flow through the offgas system was 0.0062 t^3/min/MWt points to the fact that the authors of NUREG-0016 assumed that power is proportional to offgas flow rate. GE disagrees. For a Lungmen sized plant, using the NRC NUREG-0016 factor, the resulting flow rate is seen to be 4005 x 0.0062 = 24.8 ft^3/min (42 m^3/h) which is not correct for this plant. The assumed air in-leakageis preliminarily set to 51 m^3/h (30 ft^3/min). Therefore the equation was readjusted by removing the factor 0.0062 and making the power a constant and the flow rate through the offgas system a variable.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

T Track Number: 15-025

PSAR Sections: 15.2.5

Question Date: April 20, 1998

PSAR Question:

In general a turbine trip will initiate a bypass valve opening. However, low condenser vacuum will initiate a bypass valve closure. Please (1) explain the behavior of bypass valve during a loss of condenser vacuum event, (2) clarify whether a 0.15 seconds time delay for reactor scram will be assumed for the loss of condenser vacuum event, and (3) discuss the radiological consequences of the loss of condenser vacuum event based on the Lungmen specific bypass line design.

Response:

1. During a loss of condenser vacuum event, a main turbine trip and turbine stop valve closure will occur. Upon turbine trip, the bypass valves will open and will close automatically when condenser pressure increases to a prescribed value to prevent condenser pressure from going above atmospheric pressure.

2. A 0.15 seconds time delay for reactor scram will still be assumed for the loss of condenser vacuum event.

3. The radiological releases for a loss of vacuum event are bounded by the main steam line break accident (PSAR Section 15.6.4). On initiation of the turbine trip, the bypass valve will redirect steam flow directly to the turbine condenser for a period of 4 to 5 seconds. During this period the condenser will be at negative pressure and continue in a steam condensing mode. The resultant radiological releases will be minimal. After a period of 4-5 seconds, the bypass valves will shut within approximately one-half second (expected closure time based on the characteristics of a servo controlled valve) followed shortly thereafter by MSIV closing. No damage to the fuel or fission product spiking in the reactor water will occur as a result of these events. Therefore, total releases would not occur as a result of the brief bypass valve blowdown (before full closure) to the condenser during the period when the condenser pressure is increasing and as a result of SRV blowdown to the suppression pool following MSIV closure. Blowdown to the suppression pool will result in minor releases as given in subsection 15.2.4.5.3.
   In the above loss of condenser vacuum scenario, the vacuum decay rate is expected to decrease once steam inputs from the MSIV and Bypass valves have stopped after valve closures. This is most certainly bounded by the direct environmental release in the main steam line break of a 5.5 second blowdown

including water release from vessel swell to the steam lines before MSIV closure. The Loss of Condenser Vacuum analysis assumes a conservative 6.78 kPa/s vacuum decay rate, with both the bypass valves and MSIVs assumed to start closing at the same vacuum trip pressure setting.

PSAR Section 15.2.5.3.1, third paragraph, will be revised where it says, "The bypass is signaled to close at a vacuum level of about 3.38 kPa less than the stop value closure." The typographical error in pressure will be corrected to show 33.8 kPa to agree with Table 15.2-15.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-026

PSAR Sections: Ch. 15

Question Date: April 29, 1998

PSAR Question:

1. Please explain whether the following sequence of events listed in Table 15.1-7 is reasonable. Why the signal "Reactor scram and trip of 4 RIPs" is only inhibited after 0.4 seconds ?
2. Please explain that in the analysis of RIPs trip events, why consideration was given to pump flow reverse conditions ? Is it because of the assumption that the anti-reverse rotation device not functioning ? or flow reverse can really happen to tripped pumps ?
3. Please explain that during ATWS, why the initiation of SLCS and ADS mitigation would use different ATWS permissive signals of neutron monitors. The former would use SRNM ATWS permissive and the later would use APRM ATWS permissive. Is there any special considerations ?
4. Please provide the setpoint value of rapid core flow coastdown that would trip the reactor.

Response:

1. There is a 150 milli-second delay for reactor scram and 4 RIP trip during turbine trip or load rejection to confirm if the bypass valves are open. PSAR Table 15.1-7 sequence of events time for reactor scram and 4 RIPs inhibited will be changed from 3.3 (est.) to 3.0 (est.). Please also see the response to question Track No. 15-005.
2. By design each RIP has an Anti-Rotation Device (ARD) which is located at the bottom of the RIP motor and prevents a reverse rotation of the RIP. The ARD also prevents reverse rotation during normal plant operation when one RIP is stopped and the other RIPs are operating. However, even though a stopped RIP will not have reverse rotation, it still has reverse water flow through it, when other RIPs are in operation. As shown in PSAR Table 15.3-1, "Three RIP s Trip," and Table 15.3-5, "One RIP Seizure," the pump flow reverses through the stopped pumps.
3. The Neutron Monitoring System ATWS permissive signal from the SRNM Subsystem is sent to the Safety System and Logic Control (SSLC) to initiate the automatic boron injection function of the SLC and the feedwater pumps runback function of the Feedwater Control System.

   However, the ATWS permissive signal from the APRM Subsystem is sent to the Automatic Depressurization System (ADS) of Main Steam System (within the SSLC) to control the ATWS auto-ADS inhibit function.

During an ATWS event, the SLC auto initiation and ADS mitigation use different ATWS permissive signals of the Neutron Monitoring System (NMS) for signal diversity.

4. The core flow rapid coast-down scram trip is initiated upon the condition of large negative change in reactor vessel core flow when the initial reactor power is greater than or equal to 80% of Nuclear Boiler Rated.
   The scram initiation from rapid flow coastdown is given by the following expression when $Z(t) < 0$.

$$Z(t) = F(t) - A * F(t - T) + B$$

where:

F(t) = Core flow as measured by core plate differential pressure at time t (% of rated core flow)

T = Predefined delay time in seconds (typically 3 seconds)

F(t - T) = Core flow as measured by core plate differential pressure at time of (t - T) (% of rated core flow)

A, B = Predefined constants.

The scram trip setting of the rapid flow coastdown scram trip is dependent on the delay time T and the constants A and B. The constants A and B will be determined analytically and will be provided in the FSAR.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-027

PSAR Sections: 15E

Question Date: April 29, 1998

PSAR Question:

1. It is mentioned in 9.3.5.3 that two sets of the component required to actuate SLC are provided to assure the availability. It is also mentioned in 15E.6 that the operation of both pumps .... Please clarify how many pump in operation is assumed in 15E ATWS analysis.
2. It is mentioned that the boron would reach core 60 seconds after the initiation. Is this 60 seconds time delay based on a single pump operation or both? Does this 60 seconds include the time required to build up the boron concentration gradient within HPCF pipe?
3. Please provide the sequence of events for Fig. 15E-2 through Fig. 15E-14.

Response:

1. The ATWS analysis was performed with two SLC pumps operational. The total boron injection rate is 22.7 m^3 /h (100 GPM).
2. The 60-second delay is based on the assumption that both SLC pumps are operating. This considers the pump startup delay and building up of the boron gradient in HPCF pipes.
3. A representative bounding ATWS sequence of event for All MSIV Closure and Boron Injection is listed below.

Event	Time (sec)
Main steamline isolation initiated	0.0
Scram from MSIV position signal fails	0.3
Scram from high flux signal fails	1.2
Scram from high pressure signal fails`	2.1
Relief valves start to lift, ATWS high pressure setpoint reached (7.76 MPaG), non-M/G set RIP tripped, start ARI, start FMCRD, logic initiated for feedwater runback (2 minutes timer) and boron injection (3 minutes timer)	2.3
Main steamline isolation completed	3.0
Rod motion due to FMCRD expected to start - FAILS	3.3
Rod motion due to ARI expected to start - FAILS	18
SRNM shows power high, feedwater runback started	123
Reactor water level reached Level-2. Initiate high pressure makeup water injection. (Note)	164
SRNM shows power high, boron injection pump started	183
Boron reaches core after 60 seconds transportation delay	243
Suppression Pool Cooling begins	660

Hot shutdown achieved	19.4 minutes

Note : HPCF injection logic was initiated at Level-2 in the analysis. This assumption yields a more conservative suppression pool temperature than would the normal Level-1.5 HPCF initiation. Both RCIC and HPCF are initiated at Level-2 in the analysis.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-028

PSAR Sections: 15.7.4 and 15.7.5

Question Date: May 22, 1998

PSAR Question:

1. Inconsistent number of failed rods in fuel handling accident: 172 rods on page 15.7-6 and 115 rods on page 15.7-13.

2. Non-conservative average burnup assumed for failed fuel, 32 GW d/t. (Table 15.7-8 and 15.7-12) TPC is considering up to 43 GW d/t burnup.

3. Non-conservative assumptions of fuel bundles in one cask, 18 in Table 15.7-12. For the current spent fuel cask over 100 MT, total number of bundles could exceed 88 BWR fuel bundles.

Response:

1. An attempt was made to show the effects of a fuel handling accident event for GE12 fuel design whereas the SSAR only discussed the GE6/7 (8x8) fuel design, since no new analysis was performed for the PSAR. In the SSAR analysis of a fuel assembly dropped into the core, a GE6/7 bundle weight of 279.87 kg was used along with an older fuel handling equipment grapple component design weight of 158.76 kg in a postulated drop of 13.4 meters resulted in the failure of 155 fuel rods. As stated in PSAR Subsection 15.7.4.3.3, an analysis with the new GE12 fuel was performed in Reference 15.7-1, GESTAR III Republic of China, General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A-7-RC, August 1995, section 10.5.1.5. In this analysis of a GE 12 bundle which weighs 302 kg, the fuel bundle along with a mast and grapple head weight of 280.8 kg was assumed to drop from a height of 10.4 meters, resulted in the failure of 172 failed fuel rods. PSAR Table 15.7-8, Fuel Handling Parameters has a footnote designating that the table is for the Standard ABWR design. PSAR Subsection 15.7.4.3.3 will be revised to read as follows:

15.7.4.3.3 Results

Because of the complex nature of the impact and the resulting damage to fuel assembly components, a rigorous prediction of the number of failed fuel rods is not possible. For this reason, a simplified energy approach was taken and numerous conservative assumptions were made to assure a conservative estimate of the number of failed rods.

The number of failed fuel rods (8x8 fuel) for the Standard ABWR was determined by balancing the energy of the dropped assemblage against the energy required to fail a rod. The wet weight of the dropped bundle is 279.87kg and the wet weight of the grapple component is 158.76kg. The drop distance is 13.38 m. The total energy to be dissipated by the first impact consist of one half of the energy considered to be absorbed by the falling assembly and one half by the impacted assemblies.

No energy was considered to be absorbed by the fuel pellets (i.e., the energy was absorbed entirely by the non-fuel components of the assemblies). The dropped assembly was considered to impact at a small angle, subjecting all the fuel rods in the dropped assembly to bending moments. The fuel rods are expected to absorb little energy prior to failure as a result of bending. For this reason, it was assumed that all the rods in the dropped assembly fail. The total number of failed rods on initial impact which consist of the number of rods failed in the impacted assemblies plus the number of rods failed in the dropped assembly was 106.

The assembly was next assumed to tip over and impact horizontally on the top of the core. The remaining available energy was used to predict the number of additional rod failures. The available energy was calculated by assuming a linear weight distribution in the assembly with a point load at the top of the assembly to represent the fuel grapple weight.

As before, the energy was considered to be absorbed equally by the falling assembly and the impacted assemblies. This energy available deforms the clad in the impacted assemblies and the number of additional failures in the impacted assemblies was calculated to be 9 fuel rods.

Since the rods in the dropped assembly were considered to have failed in the initial impact, the total failed rods in both impacts is 106 + 9 = 115.

For comparison, using a GE12 (10x10) fuel rod array, a similar analysis methodology resulted in 172 failed fuel rods. See analysis section 10.5.1.5 of Reference 15.7-1. The fuel handling accident analysis with GE12 core design with the Lungmen fuel handling equipment will be presented in the FSAR.

2. The inventory used for the fuel handling accident is an equilibrium core inventory for a 29 GWD/t core average burn up with peak rod burn up of 35 GWD/t. This inventory is conservative for radiological analysis of the fuel handling accident when compared to an equivalent inventory with peak burn up of 43 GWD/t. The reason for this involves consideration of those radionuclides which are significant for offsite dose calculations and their production and burn up in a reactor core. With the exception of Kr-85, which is radiologically insignificant in comparison to the other isotopes, all the five iodine isotopes and 13 noble gas isotopes reach equilibrium in a core within ninety days of beginning operation. The equilibrium concentration of these isotopes is directly dependent on the local power density and only somewhat dependent on the fissile nuclide inventory. As a bundle is burned through its lifetime, this weak dependency on fissile nuclide inventory combined with proper bundle management, with bundles being moved from high power density regions of the core into lower power density regions, results in the isotopes going through a maximum density toward the end of the first cycle in the core. Following this, the density of these radiological significant isotopes slowly decreases over the lifetime of the bundle. By using an equilibrium core with a lower peak burn-up, the overall inventory for these isotopes is slightly over estimated and the resulting calculation slightly conservative. Since the core average inventory is used in radiological calculations, the core average inventory for the 35 GWD/t is larger than a 43 GWD/t inventory and as such is conservative.

3. For the Standard ABWR design, the spent fuel cask drop accident event was based on a cask design that contained 18 spent fuel bundles. This shipping cask design was similar to the US NRC licensed IF-300 shipping cast used in the domestic Shoreham plant fuel transfer project. For the Lungmen NPS, the spent fuel cask and design will be supplied by others, and once its final design capacity is determined, new analysis results will be provided in the FSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-029

PSAR Sections: Ch. 15

Question Date: May 28, 1998

PSAR Question:

1. The following statements were given in NUREG 1503 page 15-3 "GE initially appropriately categorized the inadvertent RHR shutdown cooling operation event as an accident rather than an AOO. Which was a significant deviation from the SRP. GE recategorized this event as a moderate frequency event (an AOO) and applied the appropriate acceptance criteria in the SSAR". However, this event is still categorized as a limiting fault as shown in page 15.1-12 of PSAR. Please clarify it.
2. As shown in page 15B-11 of PSAR, the energy of impeller missile is 0.09MN • m, the critical energy of RPV shell is 9.41 MN • m. However, the following statement is also given "... the impeller missile KE is approximately ... one-tenth the RPV shell CE". Please correct the inconsistency.
3. In Figure 15A-51, why it does not show the scram signal caused due to turbine stop valve closure but inhibited due to bypass valves fast opening?
4. In PSAR, it was stated that reactor will automatically scram during big earthquakes. But for earthquakes not large enough to cause reactor scram, will they affect those non-safety systems? Please review the earthquake caused events and their cause of failures of related non-safety systems to the analysis in Chapter 15. Following are some examples:
(1) Rod Withdrawal Error at Power
PSAR indicated that this it is not necessary to analyze this accident because of ATLM and multi-channel rod block monitor but both of them are not safety systems.
(2) Fast closure of one turbine control valve
If fast closure of one turbine control valve was caused by earthquake, then the bypass valve open initiated by SBPC could possibly fail too due to earthquake.

Response:

1. PSAR Section 15.1.6.1.2 Frequency Classification, indicates the event hould be" categorized as a limiting fault, and the second sentence states, riteria for moderate frequency incidents are conservatively applied." That is to say that the classification is moderate frequency.

2. The fourth paragraph of Subsection 15B.3.4.1 will be revised as follows:

   15B.3.4.1 Missile Generation (4th paragraph)
   Comparing the information above, the impeller missile KE is approximately one-half the shroud CE and one one-hundredth the RPV shell CE.
3. Figure 15A-51 is for the standard ABWR design. This figure and some other Chapter 15A figures will be modified to reflect the Lungmen specific design in the FSAR.
4. Generally speaking, all safety-related structures, equipment and components are designed to remain functional within applicable stress and deformation limits when subject to the vibratory motion of the Operating Basis Earthquake (OBE) in combination with normal condition loads. Those structures, systems, and components necessary to assure the integrity of the reactor coolant boundary, the capability to shutdown and cool down reactor and maintain it in a safe shutdown condition, and prevent or mitigate the consequences of accidents that could result in potential offsite exposures greater than 10CFR Part100, are designed to remain functional during the Safe Shutdown Earthquake (SSE).
   On the other hand, non-safety related equipment is not required to remain functional during an OBE or SSE event. Design Basis Transient events (Safety Limit MCPR, Minimum Water Level, etc.) make no assumption of a specific transient concurrent with an earthquake in intensity up to the OBE or SSE (multiple failures). Some transient events can be made worse concurrent with a seismic event, however, in all cases intervention by the control room operators, in conjunction with the minimum required safety-related structures, equipment, and components, ensure that the reactor can be safely shutdown and cooled down and maintain in a safe shutdown condition.
   Please refer to PSAR Section 3.7.2.8 and note f in Table 3.2-1d (Page 3.2-68) for safeguarding safety-related systems against failure of the non-safety related systems (Seismic Category IIA, IIB, or IIC). This assures that failure of non-safety related systems would not interfere with the function of the safety-related systems under the full range of earthquake levels.
   Rod Withdrawal Error (RWE) at Power - The Multi-Rod Block Monitor (MRBM) and the Automatic Thermal Limit Monitor (ATLM) are non-safety related systems, which can fail in a seismic event. The RWE at power transient is assumed to result from either a procedural error by the operator or from the malfunction of the automated rod withdrawal control logic in which a gang of control rods is withdrawn continuously. During a seismic event with failure of both the MRBM and ATLM, the control room operator can take manual control to insert controls and/or reduce core flow if an operating limit is reached, or manually scram the reactor. It is not considered credible that the operators would try to withdraw control rods during a seismic event, so the functions of the MRBM and the ATLM are not required in such a situation.
   Single Turbine Control Valve Failure with Bypass Valve Failure - For the postulated event of an earthquake generated fast closure of a single turbine

control valve in conjunction with a failure of SBPC (bypass valves fail to open), no load reject scram signal is generated. The reactor may scram on high neutron flux or high reactor pressure as a result of the transient event.

ROCAEC Review Comment:

According to your response on part 4, one of the RWE root cause is malfunction of the automated rod withdrawal control logic in which a gang of control rods is withdrawn continuously. However, the failure of digital I&C system will not result in just "function" or "not function". It may cause the system to react in the opposite direction. Our further comments are:

1) Can earthquake be the root cause of RWE?
2) Is it probable that MRBM and ATLM fail at the same time when RWE?
3) If yes, could fuel rod be damaged before the operator can manually insert the control rods to safely shutdown the reactor?

Further Clarification:

For this scenario of multi-equipment failures concurrent with an earthquake below the automatic RPS scram setting, it is expected that the control room operator still has the ability to reduce reactor power by rod movement, reduce recirculation flow, or manually scram the reactor. Generally speaking both safety-related equipment and non-safety-related equipment will fail into a safe state.

1) It is highly improbable that an earthquake of intensity below the RPS high seismic activity scram setting, would be the root cause of a rod withdrawal error.
2) The MRBM and ATLM are diverse systems with diverse software and hardware. Earthquake caused failures of the MRBM and the ATLM concurrent with RWE are highly improbable, but possible as was mentioned in the original response. Both the MRBM and the ATLM are independent and diverse dual channel systems, which interface with the Rod Action and Position Information (RAPI) Subsystem of the Rod Control and Information System (RCIS). The RAPI subsystem includes verification logic and enforces rod blocks based upon diverse signals both internal and external to RCIS. External input signals to each RAPI channel that are used for rod block logic originate from several sources including the MRBM. Because of this high level of diversity and redundancy, it is concluded that an earthquake induced RWE is highly unlikely.
3) As this scenario is multiple failures of diverse non-safety related systems concurrent with an earthquake, it is expected that the control room operator will have time and the necessary information from displays to intervene to safely shutdown the reactor. On the other hand for beyond design basis large earthquakes, core damage probability is assessed in the Probability Risk Analysis (PRA), Appendix A of the PSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-030

PSAR Sections: Ch. 15

Question Date: May 28, 1998

PSAR Question:

1. The scram time shown in Table 15.0-5 is less than that in Table 15.0-6 of SSAR. Please explain the reason.
2. The scram time shown in Table 15.0-5 is quite different from that of 4.6.1.2.5.3. Is that to fulfill the 0.8 conservatism factor in SRP? Please explain.
3. It seems that the scram reactivity curve shown on Table 15.0-4 is based on GE 6/7 fuel. What would the curve be if GE12 is the fuel loaded? And, what would be the effect of ΔCPR calculation of load rejection with bypass failure.

Response:

1. The ABWR reference design scram time requirements shown below for SSAR Table 15.0-6 were derived from the results of both tests and analyses based on the scram times of a BWR-6 boron carbide control blade weight and a hafnium control blade weight. PSAR Table 15.0-5 scram times requirements were derived from the scram times of a BWR-6 boron carbide control blade only.

	SSAR Table 15.0-6	PSAR Table 15.0-5
Rod Insertion (%)	Scram Times	Scram Times
10	0.46	0.46
40	1.208	1.20
60	1.727	1.71
100	3.719	3.70

For the reference ABWR testing program, with "Full Load Rejection without Bypass Pressure Transient (vessel bottom pressure 8.6 MPa maximum)", the following scram times for both the boron carbide and hafnium only control blades were derived:

	BWR-6 Blade (B_4C)	Hafnium Blade
Rod Insertion (%)	Scram Times	Scram Times
10	0.46	0.46
40	1.20	1.32
60	1.71	1.98
100	3.70	4.00

The FMCRD was designed to be capable of performing with expected to be heavier hafnium-type control blades to allow for the capability to use hafnium blades at either the control cell core locations only or throughout the entire core. The scram times requirements imposed on the hafnium blades were derived during earlier phases of the ABWR program based on estimates of the FMCRD's mechanical capabilities with an assumed blade weight of 140 kilograms and hafnium blades only in the control cell core locations. For this core there were 192 B_4C control blades and 13 Hafnium control blades. The SSAR scram times were derived, for example at 40% control rod insertion, as follows:

$$192/205*1.20 \text{ sec} + 13/205*1.32 \text{ sec} = 1.208 \text{ sec.}$$

For Lungmen, there will be two types of control rods, the Duralife 230 (B4C + Hafnium) used for power shaping (45 total) and the Duralife 120 (B4C only), used in shutdown positions in the reactor (160 total) not used for power shaping. The maximum weight specified for any control blade type for the Lungmen core will be 104 kilograms which is similar in weight to the BWR-6 type control blade and its scram time was shown in the PSAR table.

2. The scram times shown in Table 15.0-5 are for the fast pressurization transient analyses. PSAR Section 4.6.1.2.5.3 scram times are for steady state conditions (vessel pressure below 7.48 MPaG). The 0.8 conservatism factor in the SRP is considered in the scram reactivity curve and not considered in the scram times.

3.

3. The scram reactivity data shown in Table 15.0-4 is only used for the REDYA analyses. The PSAR analysis for ΔCPR calculation of load rejection with bypass failure was performed with the ODYNM code. Table 15.0-4 is based on the Standard ABWR GE6/7 fuel design. The Lungmen specific scram reactivity curves will be provided in the FSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-031

PSAR Sections: Ch. 15

Question Date: May 28, 1998

PSAR Question:

1. Generally speaking, fuel failure is assumed for all rods of which the CPR falls below MCPR safety limit. However, special criteria with 600°C/60 sec are raised in 15.3.1.5.2. Please explain the basis of the criteria.

2. Regarding the question aforementioned, can the criteria applied directly to GE12 fuel design without any modification? Please explain the reason.

3. Using Chinshan unit 1 cycle 17 reload design as an example, almost all the rods will have exposure more than 20 GWD/MTU before EOC of 2nd cycle. It seems that the statement given in 15.3.1.5.2 "In general, fuel rods... 20 GWD/MTU... for more than two fuel cycles" can not be applied generally. Please provide further explanation.

Response:

1. The US NRC classified the Trip of all RIPs and Pressure Regulator Downscale Failure in the special category of anticipated transients involving a common-mode software failure and established a special acceptance criterion for the radiological dose calculation for these two events. The US NRC does not require that fuel failure be assumed in dose calculations for fuel rods that are under approximately 600 °C for less than 60 seconds. This time and temperature criterion is based on test data for fuel that has achieved up to 20 GWD/MtU burnup; thus, it may be applied only to fuel with burnup of less than 20 GWD/MtU.

   It is possible for a reactor core to have more than 0.1% of the total rods subject to boiling transition if there is one or more fuel rods lower than the MCPR safety limit. Fuel failure is not assumed for all rods of which the CPR falls below the MCPR safety limit.

2. The above criteria are based on the test data for fuel that has achieved up to 20 GWD/MTU burnup, and it should be able to apply to the GE12 fuel design. We will further evaluate these criteria when we perform the FSAR radiological calculations.

3. The original SSAR fuel cycle was based on an energy utilization plan (EUP) that is lower than the Lungmen's. Fuel rods will have exposure more than 20

GWD/MTU before EOC of the 2nd cycle with the Lungmen EUP. PSAR Subsection 15.3.1.5.2 will be modified as follows:

15.3.1.5.2 Trip of All Reactor Internal Pumps (last paragraph)

In general, fuel rods with more than 20 GWd/MTU exposure are those remaining in the core for two or more fuel cycles. In the equilibrium cycle, these fuel bundles account for about 50% to 75% of the total bundles. The power generated by these bundles is usually 20% less than that of the hottest bundles. On a trip of all RIPs event, less than 0.2% of the fuel rods get into boiling transition. Therefore, the requirements of the 10% of 10CFR100 are met.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-032

PSAR Sections: Ch. 15

Question Date: May 28, 1998

PSAR Question:

1. Please explain why the loss of offsite power (Loop) is not included in the accident analysis of Lungmen?

2. For Lungmen design, when loss of normal preferred offsite power, PG, PIP and IE buses will be transferred to alternate preferred offsite power source. Reactor scram signal caused by generator and high voltage breakers opening may be inhibited due to bypass valves fast opening. However, if alternate preferred offsite power was lost as well, will it immediately generate a scram signal? Please explain.

Response:

1. For the Lungmen design with 110% turbine bypass valve capacity, the loss of the normal preferred offsite power and alternate preferred offsite power will be similar to the generator load rejection event. Loss of the offsite power will generate a turbine control valve (TCV) fast closure and if the turbine bypass valves function properly, the reactor scram and recirculation pump trips will be inhibited. Since the scram is inhibited, with the main generator output breaker remaining closed, the turbine generator continues to supply power (approximately 2-3% steam flow through the TCV) to the auxiliary plant loads, with nothing of consequences occurring. This event is bounded by the load rejection without bypass event, therefore, not analyzed for the PSAR. It will be analyzed for the FSAR. The following change to the text will be made:

   15.2.6.1.1.2 Loss of Grid Connections (second paragraph)

   Should this occur, it would result in the same sequence of events as described in Subsection 15.2.6.1.1.1 for the Standard ABWR design. Analysis for the Lungmen NPS design will be provided in the FSAR.

2. In the event that the main generator breaker also opens in addition to the assumption that both power sources are lost, the loss of the grid will generate a turbine control valve (TCV) fast closure and a turbine trip by the generator protection logic. The turbine bypass valves will open using its own hydraulic accumulator and will modulate the bypass valves for at least six seconds, responding to control pressure only. The reactor scram will be inhibited due to

bypass valves opening. If no other direct scram (e.g. low water level 3 scram) has occurred before the bypass valves close, a direct scram on insufficient opening of the turbine bypass valves will be initiated by the RPS logic prior to significant pressurization occurring (if reactor power is above 40%). However, a low reactor water Level 3 scram may occur first because the condensate pumps will trip upon the main generator breaker opening. Condenser pressure will also increase due to tripping of the circulating water pumps. This may cause a turbine trip and scram depending on how fast condenser pressure increases as compared to the transient change of reactor water level.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-033

PSAR Sections: Ch. 15

Question Date: May 28, 1998

PSAR Question:

1. According to PSAR Section 10.4.7.2.2, two/three feedwater pumps normally operate in parallel. Please clarify indeed how many feedwater pumps will operate normally?

2. If three feedwater pumps normally operate in parallel, please justify the assumption, 130% of rated total feedwater flow, used in the analysis of the feedwater controller failure-maximum demand event.

Response:

1. At the time of submittal of the PSAR, the number of steam driven feed pumps operating at rated power conditions had not been decided. The analysis was based on two feedwater pumps in operation. Analysis for the FSAR will be based on actual designed number of pumps operating at rated power conditions.

2. The PSAR analysis was based on two pumps in operation at rated power conditions. Total feedwater flow will not exceed 130% rated for the feedwater controller failure-maximum demand event.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **15-034**

PSAR Sections: **Ch. 15**

Question Date: **June 10, 1998**

PSAR Question:

1. It can be seen from Fig. 4.4-1 that the core flow is only 20% under natural circulation condition. However, it is around 30% for Kuosheng NPP which is a BWR-6 design.

   (1) It is shown on PSAR Fig. 15.3-2 that the core flow will drop to around 30% under all pump trip. Is this consistent with the RIP design?

   (2) What is the core flow rate adopted in the Lungmen accident analysis which involves reactor operation around natural circulation regime (such as LOCA)?

2. Please clarify that the "(7) High Pressure (7.76 MPaG), and SRNM ATWS permissive for 2 minutes" for ATWS logic and setpoints on p.15E-3 is to actuate which device?

Response:

1.

   (1) The ABWR "All Pump Trip" event was analyzed for 20 second from the time trip of all RIPs is initiated. For this analysis, the pumps are conservatively simulated with minimum specified rotating inertia's. The core flow at Time =20 second is approximately 24% of rated or 21.7% of the initial (111%) . The pump is still coasting down at that time.

   (2) In the Lungmen LOCA analysis, all RIPs are assumed to trip at the start of the transient event, the core flow will decrease rapidly to approximately 20% of the initial core flow rate for the large break case. The core flow will decrease to approximately 30% for the small break case.

2. There is a typing error on page 15E-3, item (7) should be a bullet item associated with item (6) Feedwater runback and item (8) should be renumbered to item (7). Upon receipt of an ATWS trip signal from the Safety System Logic and Control (SSLC) System, the Feedwater Control (FWC) System initiates a reactor feedwater pump runback to minimum speed. This prevents dilution of the boron injected to shut down the reactor during an ATWS event. Additionally, for the ADS inhibit, the text should use the word "unless" instead

of "when". The ADS Permissive logic is "Normally Inhibited", unless both power and water level (level 1.5) are below their setpoint. The use of the word "unless" is consistent with the wording in PSAR Chapter 7 and 16. The PSAR will be modified as follows:

15E.4 ATWS Logic and Setpoints

(6) Feedwater runback

- High pressure (7.76 MPaG) and SRNM ATWS permissive for 2 minutes.

(7) ADS inhibit

- Automatic initiation of ADS is inhibited unless there is a coincident low reactor water level signal (level 1.5) and an APRM (Average Power Range Monitor) ATWS permissive signal.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **15-035**

PSAR Sections: **Ch. 15**

Question Date: **June, 19, 1998**

PSAR Question:

Please explain whether the effect of site terrain is taken into account in the LOCA offsite dose calculation. If yes, please explain how it was done. If no, please explain what was the assumption used in the calculation.

Response:

Lungmen Site terrain has been taken into account in the LOCA offsite dose calculation. Sixteen(16) sectors were used in modeling the terrain every fifty(50) meters from one hundred(100) out to two thousands and five hundred (2500) meters in all 16 sectors. Site terrain is an input to the PAVAN computer code used to calculate site meteorology for accident conditions. Multiple heights of terrain may be input in each sector for the meteorological calculations. For specific details see NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Material from Nuclear Power Stations", and NUREG/CR-2260, "Technical Basis for Regulatory Guide 1.145, 'Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants' ".

ROCAEC Review Comments:

(1) Please provide the dose calculation results from ground release alone and from stack release alone during LOCA.

(2) Please provide Dose Map of LOCA calculation.

(3) Please explain how the coupling effects are "combined" for clarification item 2 of the 7/20 review meeting? Also the reasonability of the description on the 0.25% conservatism?

(4) Please clarify whether the meteorological data used for stack release is taken directly from the 93m data? Or is it extrapolated to the 116m height?

(5) Please provide (a) the maximum X/Q value (900m) of stack release in LOCA dose calculation; (b) explanation whether the dose calculation in SSAR includes stack release or only roof release; (c) the impact to

Lungmen dose calculation due to the different stack heights (i.e., the current 116m and the 76m listed in SSAR)

Further Clarifications:

(1) The two points of release are the SGT stack which is an elevated release and the turbine building truck doors which is a ground level release. The doses from both these release points are given in the attached table.

	Thyroid		Whole Body	
SGT stack release:	0.43	Sv	0.0107	Sv
MSIV leakage:	1.53	Sv	0.0063	Sv
Total:	1.96	Sv	0.017	Sv

A sensitivity calculation was made by using 300m ground Chi/q values for MSIV leakage and 300m elevated Chi/q instead of 520m the maximum elevated Chi/q values for SGT stack release. The dose from both release points are given as follow:

	Thyroid		Whole Body	
SGT stack release:	0.267	Sv	0.007	Sv
MSIV leakage:	1.53	Sv	0.0063	Sv
Total:	1.797	Sv	0.0133	Sv

(2) PAVAN analysis of the meteorological data for 1993 indicate the WNW sector to be the worst for elevated release from the 116 meter elevation. The following data show the 0-2 hour Chi/q data for the WNW sector.

Distance (meter)	Chi/q Sec/m^3
500	9.27E-5
520	1.19E-4
800	1.17E-4
1000	8.92E-5
1500	5.60E-5
2000	3.91E-5
2500	3.00E-5

The 520 meter location in the WNW direction is within the site boundary. A site map is attached for reference.

(3) The PSAR uses very conservative methods for dose calculations by adding the dose from the worst elevated release in the WNW sector to the worst ground release dose. Using this method results in a total meteorological probability between 5% and 0.25% worst case.

(4) For the PSAR, the data was directly taken from the 21 m data and corrected by the PAVAN code to a height of 116 m. The 21 meter data was chosen as it was one of the two data sets found which did not contain any missing hourly data points. The 21 meter height seemed a better fit for ground application, while for the elevated release, the 21 meter data was used for consistency with the design basis meteorology, the PRA weather, and the data used to study normal release, and for the following reasons:

1) The site sets on a plane with predominate winds from the N-NE. Both the site and wind tower set in the dominate wind field without obstruction.

2) The 93 meter tower sits on a 53 meter knoll at the edge of a ridge and near other knolls which could cause some variability in measurements. Specifically the 63 meter reading could be disturbed causing a unwanted variability in stability calculations if not wind direction. Therefore until such times as study could confirm the goodness of this data it was decided not to use this data set.

For the FSAR a minimum of 2 years of elevated data will be used including any data available from the new meteorological tower.

(5) (a) The maximum X/Q value for elevated releases and ground releases are provided in PSAR Table 15.6-13.

(b) Dose calculation in SSAR only includes ground release.

(c) 因為ABWR之煙囪高度僅76m，依法規規定需視為地面排放。核四廠之煙囪設計符合高點排放要求，兩者排放機制不同，不宜比較。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-036

PSAR Sections: 15.3.1

Question Date: June 12, 1998

PSAR Question:

1. For the trip of all RIPs event, Reactor scram is initiated by the core flow rapid coastdown signal. According to T.S. This signal only work when reactor power is more than 80%.

2. When reactor power is lower than 80% and trip of all RIPs happens, core flow rapid coastdown will not scram the reactor. Vessel water level swells to Level 8, which initiates main turbine trip and Feedwater pump trip. If bypass valves opens, reactor scram will not be initiated. Reactor scram will probably be delayed until water level lowers to Level 3.

3. Has analysis been performed to assure that when reactor power is lower than 80%, trip of all RIPs is less severe than when reactor is at full power.

Response:

1. The Technical Specification has the 80% value bracketed. This is not a final value, which will be determined based on Lungmen specific analysis.

2. Agreed.

3. Analyses were previously performed for the now currently operating ABWRs by using reactor scram bypass power to be 75% of rated. Calculations showed that, for an initial power of 75% and core flow of 111% of rated, the final MCPR of All Pump Trip transient without reactor scram is slightly lower than the SLMCPR.

   The scram bypass power for Lungmen will be determined by the Lungmen specific analysis so that the special category criteria will not be violated.

ROCAEC Review Comments:

3.(a) Please clarify what does it mean for "Without Reactor Scram" in your responses concerning the analyses for Currently Operating ABWRs. Does it mean "only Core Flow Rapid Coastdown" signal will not scram the Rx or "All RPS signals are assumed not to be activated through the All RIPs trip event"?

(b) For the "All RIPs trip event" occurring at 80% power, is it worse or less severe than that occurring at 100% power? Please clarify.

(c) At what time interval will the specific analysis of All RIPs Trip event be performed for Lungmen NPS?

Further Clarifications:

3.(a) The analysis performed at 75% power and 111% flow was for the purpose of studying an effect of all RIPs trip accident on K_p, which is a multiplier for off-rated power operation applied to the operating limit MCPR (OLMCPR) at rated condition. The analysis assumes no scram through the entire event.

(b) It is possible that the event could be more severe at 80% power. The scram bypass power will be determined by a Lungmen specific analysis. Also, any core stability concern which may exist in the transient will be addressed in this analysis and the most severe results will be presented in the FSAR.

(c) This analysis for Lungmen NPS will be performed at the time when the RPS trip setpoints are finalized. It will be after the issuance of the CP, but at the start of final analyses for the FSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-037

PSAR Sections: 15.6.5.5

Question Date: June 6, 1998

PSAR Question:

Since Lungmen NPP's LPZ is quite different from the standard design, Lungmen specific calculation is important. Provide a table similar to the Table 20.3.1-1 in SSAR for Lungmen specific design. Provide a detailed description of the justification if there is any assumption used for Lungmen design deviated from which used in Table 20.3.1.

Response:

The following table is submitted to provide sensitivity studies on the LOCA analysis. It is noted that the table 20.3.1-1 from the ABWR SSAR referenced a LOCA analysis submittal based upon non-Regulatory Guide 1.3 criteria not applicable to the Lungmen PSAR. In this early ABWR submittal the following variances were made from Regulatory Guide 1.3:

- Initial release of fission products were delayed until one hour after accident initiation. This accounted for the time to decrease the water inventory to the top of core due to a line break in containment since the ABWR does not employ recirculation lines.
- After 24 hours the containment pressure was reduced by a factor of two.
- Iodine speciation was primarily particulate in nature with significantly reduced organic components.
- Suppression pool scrubbing was based upon MAAP analysis of scrubbing factors from PRA studies.

At US NRC direction, these factors were dropped or reconciled with current regulatory practice to preclude a long review process in lieu of awaiting NRC review of the then draft NUREG-1465. Since the Lungmen PSAR is based primarily on Regulatory Guide 1.3, the first three factors of this table (numeric lines 2, 3, and 4) are not applicable.

Table 15-037-1 Sensitivity Study on LOCA Analysis Parameters

		Site Boundary 2 Hr		LPZ 30 Day	
		Thyroid	*Whole Body*	*Thyroid*	*Whole Body*
1	LOCA Results	0.166	0.00355	1.96	0.0172
2	No Initial 1 Hr Hold-up	n/a	n/a	n/a	n/a
3	No Pressure Reduction at 24 hrs	n/a	n/a	n/a	n/a
4	Iodine Species Consistent with RG 1.3	n/a	n/a	n/a	n/a
5	No Suppression Pool Scrubbing	0.318	0.00447	2.83	0.0185
6	No Steamline Plateout	0.166	0.00355	1.51	0.0170
7	No Steamline Plateout or Holdup	0.166	0.00357	1.66	0.0183
8	No Condenser Plateout	0.166	0.00355	2.98	0.0178
9	No Condenser Plateout or Holdup	0.227	0.00410	13.0	0.0604

An explanation of the study components is given below.

Line 5. - A factor of two is used to account for suppression pool scrubbing. This factor is applied to the elemental and particulate species of iodine and not to the organic species at the beginning of the accident. The result is iodine releases of 12 % elemental and particulate and 1% organic to the containment atmosphere. In this study the initial iodine loadings were returned to 24% elemental and particulate and 1% organic. The results were relatively little change in whole body dose but significant increases in the thyroid doses.

Line 6. - Elemental and particulate iodines are removed by plateout mechanisms on transit through the steam and drain lines to the condenser. This process is temperature dependent and is offset by a resuspension model in which plated out iodines are chemically converted to an organic species and resuspended from the steam and drain lines. This transport model involves a time delay of approximately one hour in transit down the lines and therefore adds little to the 2 hour dose calculation as is shown in looking at line 6. With respect to the long term dose model, the thyroid dose is actually reduced. This is due primarily to the removal of the resuspension model which is extremely conservative. The resuspension model incorporates an "instant" transport model from the point of resuspension on the line to the condenser (no radioactive decay) and therefore significantly over calculates the flow of iodine to the condenser. This is shown in this case by an actual reduction in dose when the resuspension model is turned off as it is when deposition is turned off.

Line 7. - The results for this line are similar to Line 6 in that the deposition model is turned off as in Line 6 plus the reduction due to decay in transit down the steam and drain lines is also removed (about a one hour decay). Though the steam and drain line transport is disallowed, the early time dose is still dominated by the reactor building leakage pathway and so the 2 hour dose values do not vary. The 30 day values increase slightly over Line 6 due to the removal of the line delay but only slightly.

Line 8. - Condenser plateout removal only affects elemental and iodine plateout. In this case the removal factor increases only the long term iodine due to the increase in release of elemental/particulate iodine.

Line 9. - Removal of the condenser results in the single largest increase in dose since the hold-up capabilities of the condenser effects all species of iodines as well as the noble gases. With a turn over rate of 4% per day, the condenser serves as an effective mitigation system on the overall dose both at early times as well as over the full 30 day calculation.

ROCAEC Review Comments:

Please provide further explanations to the calculation results of items 6-9 of Table 15-037-1.

Further Clarifications:

The first point which must be made is that these sensitivity factors are not cumulative. That is, when no steamline plateout was considered, condenser plateout was still considered. Likewise, when no condenser plateout was considered, steamline plateout was included in the calculation.

The second point which must be included is the approximate importance of the differing iodine species to the final dose result. (Only considering thyroid dose.) The primary contribution to the total dose is organic iodines via the MSIV pathway consisting of 1.06 Sv of the 1.96 Sv thyroid dose at 30 days. Elemental and particulate iodines via the MSIV pathway add 0.0016 Sv, and resuspended iodines species which are treated as a form of organic iodines add 0.47 Sv. The total from the reactor building pathway for all species is 0.43 Sv.

The following presents a series of simplified hand calculation and estimates. The purpose of these calculations is to illustrate the most pertinent factors in developing the calculations shown in table 15-037-1.

Item 6 from the 15-037-1 table removes the equations responsible for addressing steam line plateout. Steam line plateout only affects elemental and particulate species of iodine. Therefore there is no removal in the lines but the condenser is also an effective factor in removing these species. The removal ability of the condenser is equivalent to a filter of efficiency 99.8% (Table 15.6-8 II E). Therefore though most the elemental and particulate iodines pass down the steamline, they are effectively removed in the condenser. But what this does affect is the resuspended iodines model which uses the plateout iodine as a source to create the resuspended forms. Without the plateout iodines, the resuspended contribution goes to zero and the total dose goes approximately as 1.96 - 0.47 = 1.49 which is close to the value given in Table 15-037-1.

Item 7 removes both steamline plateout and hold up in the steamlines. This is equivalent to roughly one hour of decay. Since I-131 is the dominate species in determining total dose and the half live of I-131 is 8 days, this has little effect.

Item 8 removes the condenser as an effective plateout factor. The efficiency of the condenser is 99.8% for elemental and particulate species (Table 15.6-8 II E) but not for organic or resuspended species which are treated as organic. With plateout the elemental and particulate species are treated by the simple equation

"Release = Potential Release * (1 - filter efficiency)"

With filter efficiency set to zero the elemental and particulate contribution becomes 0.0016 /(1-0.998) = 0.8. The increase shown in Table 15-037-1 was about 1 Sv which shows the correct range of values for the precision involved in the hand calculation above.

Item 9 removes both plateout and hold up. Hold up is significant with a condenser turn over rate of 3.6%/day (Table 15.6-8 II E). To explore this factor, start with the 2.98 Sv dose from the earlier study found in Table 15-037-1 and subtract the 0.43 Sv from the stack and assume this is primarily I-131 which is delivered over the time period 96 to 720 hours. Using the site meteorology, the iodine dose conversion factor of 1.08E06 Rem per curie and the breathing rate of 2.32E-4 m^3/s, the integrated release for a dose of 2.48 Sv is 2.7E8 Mbq DE I-131. With an initial inventory of 5.1E11 MBq of I-131 in the containment, it is a relatively straight forward calculation using the simple two compartment model from above to find that (ignoring all factors with respect to the steamline) that the containment must leak at an effective rate of 0.022%/day. That is, the steamline plateout and hold up effectively reduce the leakage rate seen in the condenser to a rate of 0.022% per day. Given this leakage rate of 0.022%/day, if the condenser were absent, no hold up, then the release to the environment would be a factor of 4 larger and the resultant dose would be four times larger. For a dose of 2.48 Sv, this would translate into a dose of 9.92 Sv plus the contribution from the reactor building of 0.43 Sv for a total of 10.35 Sv which is reasonably close to the 13 Sv found in Table 15-037-1.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 15-038

PSAR Sections: Ch. 15

Question Date: June 22, 1998

PSAR Question:

It is shown on Figure 15.0-1 that the maximum allowable core flow with 10 RIPs operation is 111% rated, which is much higher than BWR 6.

1. What is the initial core flow adopted in fast run out of all RIPs analysis? Why?

2. Please provide a list of the initial conditions, power and core flow, of the core-wise transients analyzed in chapter 15.

Response:

1. Initial power and flow conditions for the SSAR fast run out of all RIPs analysis were 59% NBR power and 42% core flow. These initial conditions correspond to operation at the low end of the rated control rod line, and the analysis indicated that the most severe consequences resulted at these conditions.

2. Initial conditions for system response analysis transients are provided in Lungmen PSAR Table 15.0-1. Initial conditions for transients analyzed at off rated power and flow conditions, such as run out of all RIPs will be provided in the FSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

PSAR Typing Errors

II. Chapter 15:

Table 15.7-1 Note: .....kd's taken from 1.5.2.19 and 1.5.2.20 of NUREG-0016 should be changed to "....kd's taken from 1.5.2.21 and 1.5.2.22 of NUREG-0016".

Response:

We think that no change to the PSAR is required. PSAR Table 15.7-1 Offgas System Failure Accident Parameters, Note 1 states, "Charcoal Delay calculated based upon charcoal mass using equation 1.5.1.6 of NUREG-0016 and Kd's taken from 1.5.2.19 and 1.5.2.20 of NUREG-0016." Note that NUREG-0016, Revision 1, Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from BWRs (BWR-GALE Code), Section 1.5.1.6 Charcoal Delay Systems, shows the equation to calculate holdup times using parameter K, dynamic adsorption coefficient. Parameter K is determined from Section 1.5.2.19 Dynamic Adsorption Coefficient for Krypton and Section 1.5.2.20 Dynamic Adsorption Coefficient for Xenon. Sections 1.5.2.21 Mass of Charcoal Delay System and 1.5.2.22 Detergent Waste as suggested are inappropriate sections for Note 1 of PSAR Table 15.7-1 which is referring to parameter K, dynamic adsorption coefficient.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **N-16-001**

PSAR Sections: **16**

Question Date: **January 12, 1998**

PSAR Question:

Based on 10CFR50.36a, the applicant has to submit Technical Specifications (T.S.) on effluents from nuclear power reactors. Currently, TPC has not submitted this information in the recent preliminary Technical Specifications submittal. Please submit the above information accordingly.

PSAR Response:

The control of radioactive effluents from nuclear power reactors will be covered by a program to be defined in the Lungmen NPS Offsite Dose Calculation Manual (ODCM). During the previous ABWR certification process, the USNRC had agreed with the removal of the Technical Specification section dealing with radioactive effluents from Chapter 16 of the ABWR SSAR.

The ODCM, as defined in 16.5.4.2.1, will contain the program necessary to control the radioactive effluents and maintain radiation doses to the public as low as reasonably achievable. The use of the ODCM to contain the radiological effluent program is consistent with a general trend among utilities, such as Limerick Generating Station, to consolidate information pertaining to radiation control of radiological effluents and offsite doses into a stand-alone document.

Therefore, TPC, with the assistance of GENE San Jose, will submit the information to the ROC-AEC concerning the control of radioactive effluents from Lungmen Nuclear Power Station when the Lungmen NPS ODCM is developed.

No change to the Lungmen NPS PSAR will be made as a result of the response to the PSAR question as stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **N-16-002**

PSAR Sections: **16.1.1**

Question Date: **January 12, 1998**

PSAR Question:

Should some terminology, such as MFLPD and La, be included in Section 16.1.1?

PSAR Response:

MFLPD is not specified to be the parameter for the power distribution limits in accordance with the proposed Specification 16.3.2. This is consistent with the certified ABWR Technical Specification, i.e., Chapter 16 of ABWR SSAR.

Additionally, the term "La" is defined in 10CFR50 Appendix J, which is referenced by the Surveillance Requirement, SR 3.6.1.1.1, in Specification 16.3.6.1.1. Appendix J defines the term "La" to be the "maximum allowable leakage rate", which is 0.5% of primary containment air weight per day at the calculated peak containment pressure of Pa. in accordance with SR 3.6.1.1.1.

Therefore, it is not necessary to include both terms, i.e., MFLPD and La, in Section 16.1.1.

No changes to the PSAR will be made as a result of the response to the PSAR question as stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **N-16-003**

PSAR Sections: **Section 16.3.2**

Question Date: **December 8, 1997**

PSAR Question:

In Section 16.3.2, the rules on Power Distribution Limits, only the APLHGR in 16.3.2.1 and MCPR in 16.3.2.2 were listed but NUREG-1434 covers also LHGR and APRM in 16.3.2. Please clarify why LHGR and APRM are not needed for Lungmen.

PSAR Response:

The LHGR specification was specified in the ABWR SSAR Chapter 16 for non-GE fuel only. During the ABWR certification process, the USNRC had approved the removal of the LHGR specification for GE fuel in accordance with Amendment 19 to GE Report NEDE-24011-P-A (GESTAR-II). Since Lungmen NPS will only be loaded with GE12 fuels for the first cycle, the LHGR specification is not needed in Section 16.3.2.

No credit is taken for the flow biased rod block for plants such as Lungmen NPS that have upgraded to the SAFER/GESTR LOCA basis or have implemented extended operating domain options. The flow biased scram setdown requirement under high peaking conditions eliminates the need to perform extensive analyses at off-rated conditions. Nevertheless, Lungmen NPS will perform extensive evaluations in order to implement the extended operating domain and, therefore, no credit for the flow biased scram is taken for the analyzed event which is initiated from off-rated conditions. Thus, the APRM gain and setpoints specifications are not needed in Section 16.3.2.

No change to the Lungmen NPS PSAR will be made as a result of the response to the PSAR question as stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-16-004

PSAR Sections: Section 16.2

Question Date: November 25, 1997

PSAR Question:

In Section 16.2.2, please add the following: (5) Operation of the unit shall not be resumed until authorized by the ROC-AEC.

PSAR Response:

The change will be incorporated into the PSAR. Specifically, in PSAR Section 16.2.2, page 16.2-1, the requirement "(5) Operation of the unit shall not be resumed until authorized by the ROC-AEC." will be added as an additional action to be followed with any safety limit violation.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **N-16-005**

PSAR Sections: **Section 16.3.4.6/16.3.4.9**

Question Date: **December 9, 1997**

PSAR Question:

1. The period of SR 3.4.6.2 is 31 days which is different from ITS (7 days). Please explain.
2.
3. In Section 16.3.4.9, it was not required to check the temperature difference between Bottom Head and reactor water before the startup of RIP (see BWR-6 ITS SR 3.4.11.3). Please explain.

PSAR Response:

1. In SR 3.4.6.2, the frequency of verifying reactor coolant DOSE EQUIVALENT I-131 specific activity is 31 days. This is consistent with the frequency specified for the same surveillance requirement in SR 3.4.8.2 of the BWR/6, STS (or ITS). Note that the period of 7 days is specified for verifying reactor coolant gross specific activity in PSAR SR 3.4.6.1.

   No change to the Lungmen NPS PSAR will be made as a result of the response to the PSAR question as stated above.

2. Verifying the temperature difference between Bottom Head and reactor water is not a safety requirement, so no limitation of this type is included in PSAR Section 16.3.4.9. However, this limitation is implemented in the RIP startup logic to minimize thermal shock to the reactor vessel as discussed in the PSAR Section 7.7.1.3.(5). The RFC system will prevent startup of an idle RIP if the temperature of the bottom head coolant is not within 80°C of the saturation water temperature corresponding to the steam dome pressure.

   No change to the Lungmen NPS PSAR will be made as a result of the response to the PSAR question as stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-16-006

PSAR Sections: Section 16.3.4.3/16.3.4.5

Question Date: December 9, 1997

PSAR Question:

1. LCO in Section 16.3.4.3 does not have "≤ 2gpm increase in unidentified LEAKAGE within the previous [4] hours in Mode 1" which is different from ITS. Please explain.

2. In Section 16.3.4.5:

   (1) LCO:
   [1] total LEAKAGE rate monitoring system was not included.
   [2] combination of items b and c of LCO is different from ITS Rev. 1. Please explain. (ITS is "The particulate or gaseous channel of the D/W fission product radiation monitoring system and D/W air cooler condensate flow rate monitoring system can be used.)

   (2) Action A:
   When D/W sump monitoring subsystem inoperable, should a rule be added: if other methods can be utilized to perform SR 3.4.3.1 every 8 hours to confirm the leakage rate is within limits then it is allowed to operate another 30 days - in order to be consistent with Bases.

Response:

1. LCO 16.3.4.3 in the Lungmen Technical Specifications was provided in accordance with the ABWR SSAR Chapter 16, Section 3.4.3. During the ABWR certification process, the USNRC approved the newly established leakage rate limits and the removal of 2 gpm increase limit pending on the approval of Leak-Before-Break (LBB) request from the ABWR plant applicant on a case-by-case basis. The applicant is required to prepare a plant-specific LBB analysis report and submit the report to the USNRC for approval.

However, since Lungmen does not plan on pursuing a plant specific LBB analysis, the 2 gpm unidentified leakage increase specification and the original drywell leakage rate limits specified by the BWR/6 STS (or ITS) should be used, i.e.,

a. No pressure boundary LEAKAGE;
b. Less than or equal to [19 L/min (5 gpm)] unidentified LEAKAGE;
c. Less than or equal to [114 L/min (30 gpm)] total LEAKAGE averaged over the previous 24 hour period;
d. Less than or equal to [8 L/min (2 gpm)] increase in unidentified LEAKAGE within the previous [4] hour period in MODE 1.

Therefore, LCO 16.3.4.3, Bases 16B.3.4.3 and other affected sections, such as 5.2.5.1 and 5.2.5.4 in the Lungmen NPS PSAR, will be changed accordingly as a result of the response to the PSAR question as stated above.

2.(1).[1]

The total leakage rate consists of identified and unidentified flows to the drywell equipment and floor drain sumps. The drywell equipment drain sump (LCW) monitoring subsystem monitors the identified leakage while the drywell floor drain sump (HCW) monitoring subsystem monitors the unidentified leakage. LCO 3.4.5 should include the drywell low conductivity waste (LCW) sump monitoring subsystem as part of the RCS leakage detection instrumentation.

Therefore, the following changes will be made as a result of the response to the PSAR question stated above:

a. Item a of LCO 3.4.5 will be changed to read "a. Drywell high/low conductivity waste (HCW/LCW) sump monitoring subsystem".

b. In Section 16B.3.4.5, BASES/BACKGROUND, page 16B.3.4-16, the below listed paragraph will be added after the 5th paragraph:

"The drywell LCW sump monitoring subsystem only monitors the

LEAKAGE from identified leakage sources and collected in the drywell LCW sump. This identified LEAKAGE consists of LEAKAGE from valve stem packings, RPV head flange seal, and other known leakage sources which are piped directly into the drywell LCW sump. The drywell LCW sump also has level transmitters that supply fill-rate indications in the MCR."

c. In Section 16B.3.4.5, BASES/BACKGROUND, page 16B.3.4-16, 6th paragraph; and BASES/ACTIONS A.1, 1st and 2nd paragraphs and ACTIONS C.1 and C.2 in page 16B.3.4-18, "HCW" will be changed to "HCW/LCW".

2.(1).[2]

Based on the Lungmen design, the airborne particulate channel and gaseous radioactivity channel of the drywell fission product radiation monitoring subsystem and the drywell cooling coils condensate flow monitoring subsystem as stated in Items b and c of LCO 3.4.5 in PSAR Section 16.3.4.5 are equivalent to the instrumentation stated in Items b and c of LCO 3.4.7 in BWR/6 STS (or ITS), Section 3.4.7. Thus, no change will be made to Items b and c of LCO 3.4.5 in Section 16.3.4.5 of Lungmen NPS PSAR as a result of the response to the PSAR question stated above.

Further Clarification to ROC-AEC's Comments:

The combination of RCS leakage detection instrumentation in Items b and c of LCO 3.4.5 is consistent with the ABWR, SSAR, Chapter 16, Section 3.4.5, except the names of the instruments involved in this LCO. The Lungmen specific leakage detection methods include drywell HCW and LCW sump monitoring, drywell cooler condensate flow monitoring and airborne gaseous and particulate radioactivity monitoring as described in Section 5.2.5 of the PSAR. Since the functions of these instruments are equivalent to Item a and combination of Items b and c of LCO 3.4.7 in the NUREG-1434 (or ITS), the USNRC has approved the selection of this combination of leakage detection systems for the reactor coolant pressure boundary during the previous ABWR certification process.

2.(2)

The drywell fission product radiation monitoring subsystem and/or the drywell cooling coils condensate flow rate monitor are selected in Lungmen design to indicate when and where coolant is released to the containment atmosphere. With the drywell HCW or LCW sump monitoring subsystem inoperable, no other form of sampling and/or plant instrumentation is available to provide the equivalent information to quantify leakage per SR 3.4.3.1. Therefore, it is not feasible to add the rule as stated in the PSAR question. No change will be made to Action A of LCO 3.4.5 in Section 16.3.4.5 of Lungmen NPS PSAR as a result of the response to the PSAR question stated above.

Further Clarification to ROC-AEC's Comments:

Please note that with the drywell HCW/LCW sump pump flow and sump level monitoring subsystem inoperable, protection may have been lost for the required feature's function on a component basis. Investigation and resolution of potential problem should be undertaken. Additionally, there is no other form of detection method recommended in USNRC RG 1.45 or permanently installed plant instrumentation which can provide the quantitative information required by SR 3.4.3.1, especially for those non-radioactive leaks within the drywell such as from drywell cooling system cooling flow.

In accordance with USNRC RG 1.45, the "other methods" allowed are those included in LCO 3.4.5 (other than the direct drywell HCW/LCW sump monitoring subsystem). These methods (e.g., radiation, drywell cooler condensate, etc.) allow trending to evaluate leakage. The 30 days completion time of Specification 16.3.4.5, Required Action A1 covers the time to restore the sump monitoring subsystem capabilities suggested in the system design. The 30 days completion time is reasonable because it accounts for the reliability of the multiple forms of leakage detection that are still available. The added rule suggested in the comment to this PSAR question is not practical since it may mislead and result in erroneously establishing the "time zero" at which the LCO was initially not met.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 16-007

PSAR Sections: Section 16.5

Question Date: December 23, 1997

PSAR Question:

1. Section 16.5.1.2 stated that the Shift Supervisor (SS) shall be responsible for the MCR command function. In Taiwan, the MCR responsible person is Shift Engineer (SE) so the word Shift Supervisor should be replaced by Shift Engineer. All references in other sections should be replaced too.
2. In Section 16.5.2.2 Shift Crew Composition regulations on the shift operators of the station staff, please use tabular form to facilitate reading and reference.
3. Section 16.5, Administrative Control of NUREG-1434, Rev. 0 has regulations on Training in 16.5.4 and regulations on Reviews and Audits in 16.5.5. But both were removed from Chapter 16 in Lungmen PSAR. Please incorporate those.
4. In Section 16.5.4.1, only five items were listed in the scope under procedures, programs and manuals. The Security Plan implementation and Emergency Plan implementation were missing. Please incorporate those.
5. The Review, Approval and Temporary Change of Procedures should follow NUREG-1434, Rev. 0 and should be added as well.
6. In Section 16.5.4.2, Programs and Manuals, the followings are missing: Radiation Protection Program, Process Control Program, In-plant Radiation Monitoring, Radiological Environment Monitoring Program, Pre-stressed Concrete Containment Tendon Surveillance Program, Inservice Inspection Program and Fire Protection Program. Please follow NUREG-1434 Rev. 0 and add them in.

PSAR Response:

1. To achieve consistency, the following changes will be made to the PSAR:
   a. Section 16.5.1.2, page 16.5-1, 13th line, change "shift Supervisor

(SS)" to "Shift Engineer "

b. Section 16.5.1.2, page 16.5-1, 15th line, change "SS" to "Shift Engineer"

c. Section 16.5.1.2, page 16.5-1, 18th line, change "SS" to "Shift Engineer"

2. Lungmen NPS PSAR Section 16.5.2.2 was formatted in accordance with the USNRC certified ABWR SSAR, Chapter 16, Section 5.2.2. Based on the US 10CFR50, part 50.32, the specification of Shift Crew Composition was addressed and tabulated under Section 13.1.2.3 of the Lungmen NPS PSAR, Chapter 13 in order to eliminate unnecessary repetition.
   No change to the Lungmen NPS PSAR will be made as a result of the response to the question as stated above.

3. During the ABWR certification process, the USNRC approved exceptions from NUREG-1434, Rev. 0 which included the restructuring of Section 5.0 of the Technical Specifications. "Training" and "Reviews and Audits" were identified as among the approximate 50% of other regulations in Section 5.0 of the Technical Specifications that were to be relocated to other licensee-controlled documents such as the FSAR.
   In Lungmen NPS PSAR Chapter 13, Sections 13.2 and 13.4 describe the Technical Specifications requirements regarding the "Training" and "Audits and Reviews", respectively. Detailed descriptions of "Training" and "Reviews and Audits" will be developed and provided in the FSAR for ROC-AEC's review.
   No change to the Lungmen NPS PSAR will be made as a result of the response to the question as stated above.

4. As stated above in the response to Question # 3, during the ABWR certification process, the USNRC approved exceptions from NUREG-1434, Rev. 0 which include restructuring Section 5.0 of the Technical Specifications. The "Security Plan Implementation" and "Emergency Plan Implementation" were identified at that time for relocation to other licensee-controlled documents such as the FSAR. Please note that descriptions of preliminary security plan and emergency plan have been provided in Section 13.6.2 and Appendix C

respectively in the Lungmen NPS PSAR.
TPC will provide detailed descriptions of both the security and emergency plans implementation and supply them with the FSAR submittal for ROC-AEC's review.
No change to the Lungmen NPS PSAR will be made as a result of the response to the question as stated above.

5. As stated above in the response to Question # 3, during the ABWR certification process, the USNRC approved exceptions from NUREG-1434, Rev. 0 which included restructuring Section 5.0 of the Technical Specifications. The specification regarding the "Review, Approval and Temporary Change of Procedures" was identified for relocation to other licensee-controlled documents such as the FSAR. A brief description of plant procedures was presented in Section 13.5 of the Lungmen NPS PSAR Chapter 13. Detailed descriptions of the procedures, which include the review, approval and temporary change of procedures, will be provided in the FSAR for the ROC-AEC's review.
   No change to the Lungmen NPS PSAR will be made as a result of the response to the question as stated above.

6. As stated above in the response to Question # 3, during the ABWR certification process, the USNRC approved exceptions from NUREG-1434, Rev. 0 which included restructuring Section 5.0 of the Technical Specifications. Regulations for implementing various plant programs such as the Radiation Protection Program, Process Control Program, In-plant Radiation Monitoring, Radiological Environment Monitoring Program, Pre-stressed Concrete Containment Tendon Surveillance Program, Inservice Inspection Program and Fire Protection Program were identified for relocation to other applicable licensee-controlled documents.
   Therefore, TPC, with the assistance of GE, S&W and other appropriate equipment suppliers will develop and supply the above mentioned programs for ROC-AEC review with the submittal of the Operating License (OL) application.
   No change to the Lungmen NPS PSAR will be made as a result of the response to the question as stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **16-008**

PSAR Sections: **Ch 16.3.4.7/16.3.4.8**

Question Date: **December 9, 1997**

PSAR Question:

1. In Section 16.3.4.8 LCO Note 3, please explain what the relevant safety analysis is for the 30 hours?
2. In Section 16.3.4.7 LCO, 12th line in Bases, it was said that "Operation of <u>one</u> subsystem can maintain or reduce the reactor coolant temperature as required." but in other relevant PSAR sections no mention was made that one subsystem of RHR is enough to perform the shutdown cooling function. Please explain.
3. In the Surveillance Requirement part, there was no mention of the verification of the operability of each subsystem. So how the operability of each subsystem is verified?

Response:

1. The 30 hours limit is based upon the capability of only one shutdown cooling subsystem in operation after 30 hours from initial entry into MODE 4 from MODE 3 to provide the required cooling to maintain the reactor in MODE 4. The 30 hours is conservative based upon general plant operational experience. An analysis associated with two RHR subsystems in shutdown cooling taking the plant from MODE 3 to MODE 4 within 4 hours duration has been performed. The core coolant temperature cools down from 181.1 °C to 100°C. This analysis shows that the decay heat drops sufficiently after 30 hours from the transition to MODE 4 from MODE 3 assuming maximum decay heat at the end of a fuel cycle for one RHR subsystem to maintain the plant in MODE 4.,

<u>Further Clarification to ROC-AEC's Comments:</u>

The analysis discussed above has been performed based on GE12 core design and an overall RHR heat exchanger heat transfer coefficient of 4.27 x E5 Watts/°C to evaluate the reactor cooldown rate using RHR system in both normal and emergency shutdown cooling operations. The results indicated that with two RHR shutdown cooling subsystems in emergency operation, the reactor can be cooled down to 100°C in 5.7 hours after shutdown which is well within the 36 hours requirement as described in PSAR Section 5.4.7.1.1.7. In addition, for normal operation in shutdown cooling mode, the reactor temperature can be brought down to 49°C in 39 hours after shutdown.

2. It was also shown that after 30 hours into mode 4 from mode 3, decay heat has dropped to 0.752 x E8 Btu/hr, i.e., approximately 26% of the maximum decay heat assumed at the end of a fuel cycle, which is well within the capacity of a single RHR heat exchanger (i.e., 0.905 x E8 Btu/hr). Therefore, it was concluded that one RHR shutdown cooling subsystem is capable of providing the required cooling to maintain the reactor in mode 4 condition after 30 hours from initial entry into mode 4.The sentence of "operation of one subsystem can maintain or reduce the reactor coolant temperature as required" is an optional operation practice and allows the Control Room Operating personnel to have more flexibility to maintain the reactor temperature during shutdown condition. Depending upon operating history, two RHR subsystems may be required to reduce temperature shortly after shutdown. To minimize the potential for confusion, this sentence will be changed to "Depending upon reactor decay heat load, operation of one subsystem can maintain or reduce the reactor coolant temperature as required."

3. RHR Shutdown Cooling mode operation requires an operable pump, heat exchanger and motor operated valves. The operability of RHR pump, heat exchanger and motor operated valves in the flow path are demonstrated by the satisfactory completion of the Technical Specification required pump flow test under the ECCS surveillance requirements of section 16.3.5. The operability of Shutdown Cooling primary containment isolation valves is demonstrated by the successful completion of Technical Specification required isolation testing and valve isolation time in section 16.3.6.1.3. If the RHR surveillance testing results are satisfactory and the testing frequencies are within the Technical Specification requirements, RHR Shutdown Cooling is declared operable and no additional verification is required.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **16-009**

PSAR Sections: **Ch 16.3.4.2/5.2.2.2.3.1**

Question Date: **December 9, 1997**

PSAR Question:

16.3.4.3 : The Safety function of [twelve] SRVs shall be OPERABLE.

16.3.4.3 LCO Bases: The results show that with a minimum of [eleven] S/RVs in the Safety Mode OPERABLE.....[Twelve] S/RVs are required to be OPERABLE in the Safety mode to meet single failure considerations.

5.2.2.2.3.1: The result show that only 12 SRVs are required to meet the design requirement with adequate margin.

Question(s):

1. In the safety analysis of over pressure protection of RPV, is it 11 or 12 S/RVs that assumed operable for the Safety function? It should be clarified (if it is 12, then the technical specification LCO and Bases should be changed too).

2. Section 16.3.4.3 Action A.1, 3rd line in Bases, it was mentioned that "because of additional..... satisfied with two S/RVs inoperable" which should be supported by analysis results.

PSAR Response:

1. Section "16.3.4.3" is a typo and should be changed to "Section 16.3.4.2". In addition, as described in the beginning of these preliminary Technical Specifications, information that is required to be based either upon detailed design work or other such efforts, has been bracketed, i.e., either [ ] or [XXXX] as appropriate, in the text.

   Please note that the number of required operable SRVs specified in

Section 16.3.4.2, and the minimum number of SRVs specified in Section 16B.3.4.2 to provide adequate margin to the ASME Code limit on reactor pressure during the most severe transient, were bracketed since the analysis results used to re-confirm these bracketed numbers remain to be verified during the detailed design process.

GE will perform the analysis of the over-pressure protection of the RPV for Lungmen in order to evaluate the minimum number of SRVs required for the safety function in order to provide an adequate margin to ASME Code limit on reactor pressure during the most severe transient. Then, the analysis results will be used to update affected Lungmen NPS PSAR sections, such as Sections 5.2.2.2.3.1, 16.3.4.2 and 16B.3.4.2, as required.

2. The statement quoted in the PSAR question stated above was provided (and also "bracketed") to allow the ABWR applicant's option in the pursuit of "two SRVs inoperable" licensing condition. Since Lungmen NPS is currently not planning on the two SRV inoperable condition, the analysis report will not be available to support the licensing application.

   Therefore, in Bases of Section 16B.3.4.2, Action A1, the 3rd and 4th lines, the statement "[Because of additional design margin, the ASME Code limits for the RCPB can also be satisfied with two S/RVs inoperable.]" will be removed.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 16-010

PSAR Sections: Section 16B.3.1.2

Question Date: November 25, 1997

PSAR Question:

In the last paragraph of Page 16B.3.1-6, it is mentioned that "The predicted core reactivity is calculated a 30 core simulator code as a function of cycle exposure." What does 30 mean?

PSAR Response:

"30" is a typo and should be "3D" instead.

The 3D MONICORE System uses 3D diffusion theory adapted to plant ATIP and LPRM signals to provide estimates of power distributions and thermal limits.

Therefore, the change will be incorporated into the PSAR. Specifically, in Lungmen NPS PSAR Section 16B.3.1.2, page 16B.3.1-6, 2nd line of the last paragraph, "30" will be changed to "3D".

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 16-011

PSAR Sections: Section 16.B.3.1.4

Question Date: November 25, 1997

PSAR Question:

In the section of LCO, it is mentioned that "The scram times have a margin to allow 8 of the control rods to have scram times that exceed the specified limits assuming a single stuck rod and an additional control rod failing to scram per single failure criterion." How 8, rather than other number, is reached? How to determine those eight rods in the methodology? Are the single stuck rod and the additional rod under the same hydraulic unit? If not, how is the additional rod selected?

PSAR Response:

Based on BWR/6 STS Bases of LCO 3.1.4, the scram times have a margin to allow up to 7.5% (e.g., 205 x 7.5% = 15 for the Lungmen NPS) of the control rods to have scram times that exceed the specified limits.

For conservatism, half of this allowable number, i.e., 8 (= 15/2 and round-off to the nearest integer), is established as the total number of control rods allowed to have their scram times exceeding the specified limits in the Lungmen NPS PSAR. Any operable control rods with scram times not within the specified limits are considered slow and the total number of slow rods should not exceed 8. The occurrence of a large number (e.g., more than 8) of slow control rods could be indicative of a generic control rod problem. Investigation and resolution of the potential problem should be undertaken and reactor shutdown.

A control rod is considered stuck if it could not move by FMCRD drive motor. This rod needs to be fully inserted and electrically disarmed. The other rod that was sharing the same HCU should be remain consider operational without affected by the disarmed rod.

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: **16-012**

PSAR Sections: **Section 16B.3.3.5.1**

Question Date: **November 25, 1997**

PSAR Question:

In the second paragraph of Page 16B.3.3-156, it is mentioned that "The ATLM and RWM are subsystems of the Rod Control and Information Systems (RCIS). The RCIS is a non-safety system." Please explain why RCIS is classified as non-safety system irrespective of its critical importance to reactor safety?

PSAR Response:

As stated in PSAR, Section 7.7.1.2(4) in Chapter 7, the RCIS is not classified as a safety-related system, as it has a control design basis only and is not required for the safe and orderly shutdown of the plant. A failure of the RCIS will not result in gross fuel damage. The rod block function of the RCIS, however, is important in limiting the potential consequences of a rod withdrawal error during normal plant operation. An abnormal operating transient that might result in local fuel damage is prevented by the rod block functions of the RCIS.

No change to the Lungmen NPS PSAR will be made as a result of the response to the question as stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 16-013

PSAR Sections: Section 16B.3.3.5.1

Question Date: November 25, 1997

PSAR Question:

In the last paragraph of Page 16B.3.3-158, it is mentioned that "When THERMAL POWER is above 10% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the fuel damage limit for the worst case RWE." How is the worst control rod configuration determined? And what is the methodology?

PSAR Response:

The worst cases control rod configurations of a potential RWE transient at power result either from a procedural error by the operator in which a single control rod or a gang of control rods is withdrawn continuously, or a malfunction of the automated rod withdrawal sequence control logic during automated operation in which a gang of control rods is withdrawn continuously. However, in either case, the operating thermal limits rod block function will block any further rod withdrawal when the operating limit is reached.

The methodology for Rod Withdrawal Error is described in PSAR Chapter 15.4.1 and 15.4.2.

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: **16-014**

PSAR Sections: **Section 16B.3.10.5**

Question Date: **November 25, 1997**

PSAR Question:

In the third paragraph of Page 16B.3.10-18, it is mentioned that "Since the scram function and refueling procedures and the refueling interlocks may be suspended, alternate backup protection required by this Special Operations LCO is obtained by ensuring that a five by five array of control rods, centered on the withdrawn control rod, are inserted and are incapable of being withdrawn." What is the basis for 5X5 array configuration? In addition, the word "incapable" means not allowed or inhibited? Please clarify.

PSAR Response:

The basis for the 5x5 requirement is to reduce the probability of inadvertent criticality caused by a single operator error or equipment malfunction. If the fuel bundles are removed from the cell containing the removed control rod and all the rods in the surrounding 5x5 array are inserted and disarmed, then the withdrawal of a single control rod (not in the 5x5 array) places the plant in just as safe a condition as the normal plant configuration of all rods and bundles inserted. This specification allows multiple occurrences of these 5x5 arrays (but obviously not too many given the required separation). Although specific calculations have not been performed for the Lungmen NPS configuration, pulling Group 1 control rods (which results in a evenly distributed pattern of about 1/8 of all the control rods withdrawn) during a normal startup will surely violate the 5x5 rule (if it were applicable then), and shutdown must be maintained at all times with Group 1 withdrawn. The word "incapable" in this specification means "inhibited" or "electrically disarmed", i.e., the withdrawal function of the centered rod is inhibited.

No change to the Lungmen NPS PSAR will be made as a result of the response to the question as stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 16-015

PSAR Sections: Section 16B.3.10.6

Question Date: November 25, 1997

PSAR Question:

In the second paragraph of LCO on Page 16B.3.10-22, it is mentioned that "When loading fuel into the core with multiple control rods withdrawn special reload sequences are used to ensure that reactivity additions are minimized." How are the special reload sequences determined? Please exemplify how minimal reactivity additions are achieved.

PSAR Response:

The basic procedures for minimizing reactivity additions is referring to the use of a spiral offload/reload pattern. Spiral offloading/reloading will always remove or load fuel at the periphery of the fueled region. By always adding fuel to the periphery of the loaded core with at least two water faces, you are loading fuel in a low importance region and this will minimize the worth of the cell. This is as opposed to loading in a checkerboard type fashion that could involve loading fuel into an empty cell surrounded on all sides by fueled cells. This will create a flux trap and tend to maximize the worth of the loaded cell.

TPC will develop a "special" reload sequence or procedure as required for reinstalling those control rod(s) being removed for maintenance works during refueling outage.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 16-016

PSAR Sections: Ch 16

Question Date: January 6, 1998

PSAR Question:

A time delay of 150 ms is required to verify the status of bypass valve before scram is initiated for load rejection and turbine trip. Should this timing requirement be included in the technical specification? Please explain.

PSAR Response:

The Turbine Stop Valve (TSV) Closure and Turbine Control Valve (TCV) Fast Closure provide signals to the RPS logic. A time delay is applied to these signals to determine if the required number of turbine bypass valves are opened. If there are not sufficient bypass valves opened, reactor scram and RPT are initiated. The safety analysis assumes initiation of scram and RPT will be inhibited if sufficient bypass valves are opened within 150 msecs after TSV or TCV closure is initiated (PSAR Tables 15.2-3, -4, -5, -6,-7).

The delay time of 150 msecs for scram signal initiation on TSV and TCV Closures is currently required to be tested in the Chapter 16 Technical Specifications as part of SR3.3.1.2.4 (Comprehensive Functional Test).

Therefore, no changes to the Lungmen NPS PSAR will be made as a result of the response to the PSAR question as stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 16-017

PSAR Sections: Table 16.3.3.1.1-1

Question Date: May 16, 1998

PSAR Question:

1. Item 2f : The Figure 16.3.3.1.1-1 mentioned in Oscillation Power Range Monitor is not there.

2. Item 9a : Shouldn't the Diesel Generator, RBCW, etc. be included in the "FUNCTION" section ? Please clarify. (Function 9b has included those so 9a should too)

3. Item 12 : "CRDS Water Header Charging Pressure-Low", Please clarify if there is "Time delay" design in the trip function according to PSAR 7.2.1.1.7 (7)? If there is time dalay, please provide the reason and the delay time? It should also be included in the specifications.

4. Item 24a/b : "Applicable Modes or Other Specified Conditions"; The "#" listed seems not reasonable. Please explain.

Response:

1. Figure 16.3.3.1.1-1 was inadvertently omitted in the process of generating PSAR Ch. 16 for the Lungmen NPS. Figure 16.3.3.1.1-1 referenced in the ABWR, SSAR, Ch. 16, page 3.3-17 should be used to specify the conditions at which the SSLC/OPRM instrument function is required to be operable.

   Therefore, Figure 16.3.3.1.1-1 of ABWR SSAR will be added to the end of PSAR, Chapter 16, Section 16.3.3.1.1 as a result of the response to the PSAR question stated above.

2. As stated in the Bases 16B.3.3.1.1 of PSAR , data values from four independent transmitters are used for initiating ADS A, RHR/LPFL A&C, CMS A in Function 9.a and for the isolation logic in Function 9.c. Four

additional level transmitters are used to provide data values for initiating the Diesel Generators, the RBCW, ADS B, CMS B and RHR/LPFL B in Function 9.b. Therefore, Functions 9.a and 9.b are grouped based on the instrument signals which are different than other functions listed in Table 16.3.3.1.1-1.

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

3. There is a time delay associated with the RPS trip function on CRD water charging header pressure-low as stated in PSAR, Section 7.2.1.1.7. The time delay is adjustable between 0 and 10 seconds, and is nominally set at 5 seconds. This time delay is used to avoid an inadvertent RPS trip due to CRD charging water pressure fluctuations, for example, as a result of CRD pump switch over or trip during normal plant operation. The time delay should be set fast enough so that a RPS trip is initiated before the CRD charging water pressure falls to a value which could result in slower than normal scram time.

   This time delay is required by the current Technical Specification. The surveillance requirement will be implemented by the comprehensive functional test in accordance with the SR 3.3.1.1.9.

4. This is an editorial error . As stated in Bases 16B.3.3.1.1, Functions 24a/24b are required to be operable during CORE ALTERATIONS, operations with a potential for draining the reactor vessel, and movement of irradiated fuel assemblies in the containment. A note which reads "*** During movement of irradiated fuel in the secondary containment" was inadvertently omitted during the process of generating the PSAR Chapter 16 for the Lungmen NPS.

   Therefore, the note which reads "*** During movement of irradiated fuel in the secondary containment" will be included in the table note of Table 16.3.3.1.1-1 in PSAR as a result of the response to the PSAR question stated above. In addition, the /#/ sign currently used for specifying the condition at which the Functions 24a/24b are required to be operable in Table 16.3.3.1.1-1 will also be changed to / ***/ in PSAR accordingly.

ROCAEC Review Comments:

Item 3 : Please incorporate the "Time Delay" directly into the trip function.

Item 4 : Please explain further why the records "***" that are going to be added are "During movement ...in the secondary containment" and not " ...in the containment" as in the Base ?

Further Clarifications:

Item 3 As stated in the previous response, the RPS trip function on low CRD charging header pressure has incorporated a "time delay" in the design to prevent an inadvertent RPS trip due to charging water pressure fluctuations.

Although not listed in Table 16.3.1.1-1, the "time delay" associated with any RPS function is required by the Lungmen Technical Specifications. The comprehensive functional test (SR 3.3.1.1.9) will require the simulation of time delay in accordance with the design while exercising inputs and outputs of the RPS actuation logic during the test.

Therefore, no change to the PSAR will be made as a result of this response to the Additional AEC Question stated above.

Item 4 The word "secondary containment" stated in the added note, "***", is correct and should be used accordingly in the Base 16B.3.3.1.1 for Function 24a/24b.

For consistency, the following changes will be made to the PSAR as a result of this response to Additional AEC Question stated above:

On page 16B.3.3-43, second line of the last paragraph, and its last line continued on page 16B.3.3-44, and the fifth line of the fifth paragraph on page 16B.3.3-44, change "containment" to "secondary containment".

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **16-018**

PSAR Sections: **16.3.3.1.1 and Bases**

Question Date: **May 16, 1998**

PSAR Question:

1. Please explain if a single power source failure would result in SSLC failure in two Divisions (please refer to NUREG 1503, Section 7.11, paragraph 8) for the Lungmen I&C Power Source design? Has the Technical Specification of Lungmen properly taken care of this question? Please explain.

2. Please explain the following questions related to BASES :

   (1). The numbering of Action A.2.1 ~ A.2.4 is different from the text. Please correct it.
   (2). Second line from the last of Page 16.B.3.3-15 stated "Essential 6.9KV bus....". Should this 6.9KV really be 4.16KV? (same with the 2nd paragraph of Section 16.1.3.3.3-15)
   (3). Page 16.B.3.3-15, last line of the 2nd paragraph, should LCO 3.3.1.3 be really LCO 3.3.1.4?
   (4). Do the description on "Division of Sensors bypass" in the 2nd paragraph of page 16.B.3.3-4 and the description on Sensor Channels Bypass in the 2nd paragraph of page 16.B.3.3-50 mean that all the sensors' signal of that Division will not be able to be transmitted to the four (4) downstream TLUs (SLUs) or just the TLU(SLU) of that Division? The description of the above two paragraphs on Sensors bypass seems not consistent with the 3rd paragraph of PSAR 7.2.2.2.3.1 (11). Please explain.
   (5). Following (4) above, please explain the design considerations of the case that when any Sensor Channel fails, all the Sensors of that Division will become Inoperable if Division of Sensors bypass was adopted. Please also compare that when single Sensor Channel fails, the difference in the impacts to safety functions of bypassing division or just bypassing sensor channel only.

Response:

1. Division II AC power failure will lose power supplies to Divisions II and IV battery chargers and Division II 480 VAC buses. The former are power supplies to Divisions II and IV 125 VDC buses which in turn are power supplies to Divisions II and IV vital AC buses, and the latter is the backup power supply to Divisions II and IV vital AC buses. Since SSLC is powered by vital AC buses, it will not be lost until batteries are drained out. During this period, SSLC Divisions II and IV are not disabled. Even with SSLC Divisions II and IV disabled due to single power failure, RPS will be tripped at the same time. This is true regardless what happened to SSLC since Divisions II and IV of RPS are powered from the same vital AC buses. Therefore, there is not any case which would disable SSLC and not cause a reactor scram as described in paragraph 8, Section 7.11 of NUREG-1503. Therefore, it is not necessary to include additional information pertaining to thissubject in the Lungmen Technical Specifications.

   No changes will be made to the PSAR as a result of the response to the question stated above.

2.

   (1) This is an editorial error during the process of generating PSAR Chapter 16 for the Lungmen NPS. The numbering of "Required Actions" for "Action Condition A" specified in pages 16.3.3-1 and 16.3.3-2 , i.e., A.1, A.2.1.1, A.2.1.2, A.2.2.1, and A.2.2.2, is correct and should be referenced in the Bases, Section 16B.3.3 for consistency.
   Therefore, the numbering of "Required Actions" referenced throughout pages 16B.3.3.49 to 16B.3.3.51 will be changed from "A.2.1" to "A.2.1.1", "A.2.2" to "A.2.1.2", "A.2.3" to "A.2.2.1", and "A.2.4" to "A.2.2.2" as a result of the response to the PSAR question stated above.

2.

   (2) The emergency bus voltage rating is 4.16 kV instead of 6.9 kV based on the Lungmen design.

Therefore, the voltage rating of "6.9 kV" referenced in the third line of the last paragraph on page 16B.3.3-14 and the third line of the second paragraph on page 16B.3.3-15 is in error and will be changed to 4.16 kV in PSAR as a result of the response to the PSAR question stated above.

2.

(3) The feature of RBCW/RBSW actuation on high drywell pressure, Level 1, or 4.16 kV emergency bus undervoltage is covered in LCO 3.3.1.4.
Therefore, "LCO 3.3.1.3" referenced in the last line of the second paragraph on page 16B.3.3-15 is in error and will be changed to "LCO 3.3.1.4" in PSAR for consistency as a result of the response to the PSAR question stated above.

2.

(4) Please note that the "division of sensors bypass" discussed in the second paragraph of page 16B.3.3-4 and the "channel-of-sensors bypass" referenced in the third paragraph of PSAR 7.2.2.2.3.1(11) all address the same bypass function. This bypass function disables the DTM inputs to the associated SLU and TLU in the affected division in order to permit the affected DTM to be serviced. When a division of sensors is bypassed, the sensor trip logic in all TLUs and SLUs of the affected division and the other three unaffected divisions change from 2-out-of-4 to 2-out-of-3. However, the inputs to the OLU for RPS and MSIV remain as 2-out-of-4 in all four SSLC divisions.
The "Sensor channel bypass" disables individual signal inputs from a sensor channel to the associated DTM of the affected division for the NMS only. However, all NMS bypass logic control functions are located within NMS, and none are located in RPS. There is no single NMS divisional bypass because it would affect both SRNM and the APRM at the same time.
When a sensor channel is bypassed, the bypassed sensor signals will not be transmitted to all four divisional DTMs as stated in the PSAR question above, and the sensor trip logic will also change from 2-out-of 4 to 2-out-of-3. This allows testing of sensor

channels and their inputs to the DTM.

For clarity and consistency, the following changes will be made in PSAR as a result of the response to the PSAR question stated above:

a. On page 16B.3.3-4, second paragraph, sixth line, change "all SLUs and TLUs" to "all SLUs and TLUs in the four SSLC divisions".
b. On page 16B.3.3-50, second paragraph, third line, change "for all Functions in the affected division" to "for affected Functions in all four SSLC divisions".

2.

(5) As discussed above, the "division of sensors bypass" will bypass the sensor signals on a division level and will cause the sensor trip logic of all four SSLC divisions to become 2/3. Therefore, a single failure will not result in loss of protection or cause a spurious initiation. Although the degree of redundancy is reduced, the level of safety is not compromised with the imposed LCO 3.3.1.1.

The individual "sensor channel bypass" is a special feature of the NMS system. However, the SRNM and APRM bypasses are separate input logics to RPS, each interfacing with RPS independently. All NMS bypass logic control functions are located within NMS and none are located in RPS.

If a "channel of sensor bypass" approach were adopted for SSLC, then the complexity of the SSLC system design would be increased due to multiple sensor inputs. On a practical level, little difference will occur if the bypass function is actuated at either the channel or the division level since all sensor channels in one division share the same CMU and DTM, unless this were also changed.

No changes will be made to the PSAR as a result of the response to the PSAR question stated above.

ROCAEC Review Comments:

Item 1 : Please provide further explanations regarding the following :

(1) The response did not explain the impacts to the ESF system initiation logic. Please explain.

(2) The response indicated that RPS would trip when power is lost but this is not exactly consistent with the concerns in NUREG-1503. Please clarify.

Further Clarifications:

Item 1: (1) Since ECCS function of the ESF is energized to actuate, losing SSLC DIV II due to loss of 480VAC bus coupled with either DIV I or DIV III in the bypassed position could cause ECCS failed to actuate when required. The concern addressed in NUREG-1503 would have been correct, if the "reactor scram" and the "RPS" referenced in the statement had read "ECCS actuation" and "ESF", respectively.

The isolation function of the ESF is de-energized to actuate which is similar to RPS. Therefore, the concern addressed in the NUREG-1503 will not occur as discussed in the previous response to this question.

Please note that the concern addressed in NUREG-1503 is in the second half of the paragraph. The first half of the paragraph addressed indefinite duration of bypass action in draft ABWR TS. Due to this concern, the duration of bypass action have been reduced to 30 days. This duration along with the provisions of other TS Action statements and battery backed UPS will provide adequate safety margin for reactor safety.

(2) As stated above, the concern addressed in NUREG-1503 should have been for ECCS function of the ESF. In addition, as stated in the previous response to this question, there is not any case which would disable SSLC due to losing power and not cause a reactor scram.

Therefore, no change will be made to PSAR as a result of this response to the Additional AEC Question stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 16-019

PSAR Sections: 16.3.7

Question Date: May 16, 1998

PSAR Question:

Please explain why the following safety systems were not included in the Operating Procedures ?

(1). Emergency Chiller Water System (ECW).

(2). Control Building Safety-Related Equipment Area HVAC System.

(3). RB Safety-Related Equipment HVAC System

(4). RB Safety-Related Electrical Equipment HVAC System.

(5). RB Safety-Related Diesel Generator HVAC System.

Response:

The systems listed in the PSAR question stated above are categorized as "support systems". The USNRC approved Section 16.3.7 of the US ABWR SSAR and concluded that systems included in Section 16.3.7 were adequate for the ABWR application of Technical Specifications during the certification process.

If an inoperable support system, not included in the Technical Specifications, directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported systems, it is up to the plant operator to perform additional evaluations and identify limitations. If either a loss of operability or a safety function of the supported system(s) is determined to exist, then the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

When an LCO is not met solely due to an inoperable support system that is not specified by the Technical Specifications, then the Conditions and Required Actions associated with this supported system are not required to be entered. However, other appropriate limitations and remedial or compensatory actions may be identified to be taken as a result of the support system inoperability, and an exception to entering the supported system Condition and Required Actions is made.

The above approach minimizes the number of LCOs in the Technical Specifications, thereby enhancing the flexibility of plant operations. Additionally, guidelines provided by the Safety Function Determination Program (SFDP) in Specification 16.5.4.2.12 shall be followed to assist in determining if a loss of safety function exists and that appropriate actions are identified and taken as a result of the support system inoperability.

Therefore, no changes will be made to the PSAR as a result of the response to the PSAR question stated above.

ROCAEC Review Comments:

1. The "Plant System" description in the Lungmen Technical Specification is consistent with BWR-6 ITS and GE ABWR SSAR so the Lungmen Technical Specification can be accepted in principle.

2. BWR-6 ITS, including 16.3.0.6 and SFDP (Safety Function Determination Program), etc., mainly deals with principles of actions for those supporting systems included in the specification when they become inoperable. For those supporting systems that are not included in the specification, it requires further discussion and clarification whether they can apply the same principle as stated in the response.

Further Clarifications:

1. Agreed.

   Therefore, no change will be made to PSAR as a result of this response to the Additional AEC Question stated above.

2. Agreed.

   TPC will develop an SFDP and submit the developed program with the FSAR to the ROC-AEC for review. The developed program will provide guidance to ensure that loss of safety function is detected and that appropriate actions are taken during plant operation.

The guidelines provided in specification 16.5.4.2.12 will also be evaluated for their use in determining if a loss of safety function exists when a support system not included in specifications is inoperable.

Therefore, no change will be made to PSAR as a result of this response to the Additional AEC Question stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 16-020

PSAR Sections: 16.3.3.11, 16.3.3.1.2 and Bases

Question Date: May 22, 1998

PSAR Question:

1. Please explain why radiation and neutron detectors were excluded from "Comprehensive Function Test" in the description of SR 3.3.1.1.6?

2. Please explain whether the input parameters and bypass signals of the trip functions (such as ATLM core flow rate, first stage pressure of high pressure turbine of main turbine trip bypass and openings of main turbine bypass valve, etc.) described in SR 3.3.1.1.10 should be included into the Technical Specifications.

3. Please answer the following questions related to 16.3.3.1.2 and its Bases :

   (1) Should "OPERABLE bypass" be "Operable Status" in Required Action B.3 ?

   (2) Why the Condition G Action is different from Condition C/D Action? also the relevant Bases of Condition G seems only described 3 channels inoperable but not 4 channels inoperable.

   (3) In Condition I, since RPS manual scram is diverse design and its trip logic is two out of two so if one channel is inoperable, why the Completion time of its Action is 30 days (since the function will be lost if another channel fails)? Please also explain what the Action will be if two manual scram channels are both inoperable.

   (4) Is it a typo error in the Note to condition 5 which says "a" in the "applicable Modes or other specified conditions" of Function 1.b of Table 16.3.3.1.2-1 ?

   (5) Why the Surveillance Requirements of "manual RPS scram" did not

include SR 3.3.1.2.4 (BWR-6 ITS required that logic system function test be performed every 18 months)?

4. Please answer the following questions related to 16.3.3.1.2 Bases :

   (1) The third paragraph of page 16B.3.3-70 "Three unbypassed LOGIC CHANNELS..." seemed confusing. Since Tech Spec only requires 4 channels to be operable, so the above wordings should be modified.

   (2) The third paragraph of page 16B.3.3-73 explained that when DIV II power source failed it could cause SSLC DIV II/IV to fail as well. So shouldn there be differences in the Completion time of Action when any channel of DIV II/IV becomes inoperable and when all channels of DIV II/IV become inoperable?

   (3) The text of second paragraph of page 16B.3.3-77 on Action L.1, L.2.1 and the second line from the last of L.2.2 seemed not coherent (seems words are missing). Please modify.

5. Please answer the following questions on 16.3.1.4 :

   (1) Rod Positions 10%, 40% and 60% Insertion were used in the Spec for Scram time Limit which are different from the Kuosheng T.S. (the maximum value at Kuosheng was Notch 13 or about 73% insertion). Please explain why the difference. Also some of the data used in accident analysis in Ch. 15 such as Table 15.0-5 which showed 100% Insertion time to be less than 3.7 sec. Please explain why T.S. did not specify the Scram time of 100% Insertion.

   (2) Kuosheng T.S. specified that SLOW Rod should follow the requirements of Action and check the Scram time if it meets the Spec under different notch positions and Dome pressures to make sure it is operable. But in Lungmen, this part only require to check scram time of 60% insertion and not dependent on Dome pressure. Please explain why this difference.

Response:

RESPONSES TO ROC-AEC's PSAR QUESTIONS

1. "Comprehensive Functional Test" is a set of tests that exercise each RPS, ESF, and MSIV closure function by simulating accident events that exercise the inputs and outputs of the SSLC, NMS, PRM, and RPS/MSIV actuation logic.

   Radiation and neutron detectors are excluded from comprehensive functional test surveillance requirement SR 3.3.1.1.9. because of the difficulties of simulating an accident event at the inputs of radiation and neutron detectors by a meaningful signal to exercise NMS and PRM actuation logic. However, the requirements of SR 3.3.1.1.9 for NMS and PRM are still required and can be satisfied by injecting test signals that simulate accident events at the outputs of radiation and neutron detectors and therefore exercise the outputs of NMS and PRM actuation logic.

   No changes will be made to the PSAR as a result of the response to the question stated above.

2. The total recirculation flow is specified in Chapter 16 of Lungmen PSAR since it is used to calculate the APRM flow-biased simulated thermal power monitor (TPM) setpoints and the rate of core flow decrease which initiates the RPS trip function. However, the TSV and TCV closure RPS trip bypass setpoint will be using the power level signal provided from the NMS. This modification is based on a recent Lungmen design change. This design change replaces the turbine first stage shell pressure signal with the APRMs. The APRM signal is specified in the Lungmen PSAR Ch. 16 since it is used to provide an input to initiate the RPS trip function.

   Therefore, the statements that referenced the ATLM setpoint and the statements that referenced the turbine first stage for TSV closure bypass setpoint in NOTE 2 of SR 3.3.1.1.10 in the PSAR should be changed as a result of the response to the question stated above, i.e.,

   "Automatic Thermal Limit Monitor (ATLM)", will be changed to "Thermal Power Monitor (TPM)" and "turbine first stage pressure", will be changed to "power level" in SR 3.3.1.1.10 NOTE 2 of the PSAR accordingly.

Additionally, in the Bases for SR 3.3.1.1.10 and SR 3.3.1.1.11 on page 16B.3.3-61, "ATLM" will be changed to "TPM".

3.

(1) This is an editorial error. The statement "OPERABLE bypass" in Required Action B.3 should have read "OPERABLE status" which is consistent with the statement referenced in Bases 16B.3.3.1.2 of PSAR.

Therefore, the word, "bypass", will be changed to "status" in Section 16.3.3.1.2 of the PSAR as a result of the response to the question stated above.

(2) Actions A through D addresses required actions for inoperable LOGIC channels from one channel through four channels, respectively. Actions E through G addresses required actions for inoperable OUTPUT channels from one channel through four channels as appropriate. As can be seen form the table of ACTIONS, due to different applications, the required action and completion time for OUTOUT channels are more restrictive than that of LOGIC channels with same number of channels inoperable. This is why the required actions for CONDITION G are different from those for CONDITIONS C and D although they all address three and four INOPERABLE channels.

The Bases for Required Action G.1 addressed on page 16B.3.3-75 cover both three and four inoperable channels.

No changes will be made to the PSAR as a result of the response to the question stated above.

(3) The 30 days completion time of Action I.2 is reasonable. Two hardwired manual RPS scrams which completely bypass the SSLC processing are provided. Placing the affected division in trip in accordance with Action I.1 causes the manual scram logic to

become 1/1. The automatic RPS trip actuation logic becomes 1/3 in this condition and remains available. Since the manual trip uses a minimum of equipment, there is high confidence that the manual RPS scram will be available from the remaining switch if needed. In addition, the probability of losing the automatic RPS trip logic within 30 days completion time period is quite low.

The reactor will scram automatically if the operator places two RPS divisions associated with the two inoperable manual scram channels in trip. Therefore, no further action is required after placing the affected divisions in trip in accordance with Required Action I.1, if two manual RPS scram channels become inoperable.

No changes will be made to the PSAR as a result of the response to the question stated above.

(4) The note "a" in the PSAR question stated above is an editorial error. The table note marked by "*" in Table 16.3.3.1.2-1 is applicable to the specified mode "5" for Function 1.b.

Therefore, the note "a" will be changed to "*" in PSAR for mode 5 specified for Function 1.b under the column of "Applicable Modes or Other Specified Conditions" in Table 16.3.3.1.2-1.

(5) As described above, the manual RPS scram function is totally independent of and isolated from the RPS automatic trip divisions. Please note that this function is not specifically credited in any PSAR analysis. It is retained for the overall redundancy and diversity of the RPS.

As such, comprehensive functional test, SR 3.3.1.2.4, is not required to be performed on the manual RPS trip channels. However, a CHANNEL FUNCTIONAL TEST, i.e., SR 3.3.1.2.1, is required to be performed on each manual RPS scram division at a relatively short surveillance interval of 7 days. SR 3.3.1.2.1 ensures that entire manual trip channel will operate as intended. This surveillance requirement overlaps the SR 3.3.1.2.4 testing and is performed in order to satisfy the requirements of the comprehensive functional test.

No changes will be made to the PSAR as a result of the response to the question stated above.

4.

(1) Four logic channels and output channels for the RPS actuation function are required to be operable as shown in Table 16.3.3.1.2-1.

Therefore, the statement ,"Three unbypassed LOGIC CHANNELS" in the third paragraph of page 16B.3.3-70 is confusing and will be changed to "Four LOGIC CHANNELS " in the PSAR as a result of the response to the question stated above.

(2) When Divisions II/IV become inoperable due to loss of Div II power, then Condition B is entered, and the Required Actions B.1, B.2, and B.3 should be taken within the specified completion times. This is different from losing only one division, for which the entry condition is Condition A.

No changes will be made to the PSAR as a result of the response to the question stated above.

(3) This is an editorial error. The words "L.2.2)" were inadvertently omitted during the generation of Chapter 16 of PSAR for the Lungmen NPS.

Therefore, the words "(Actions L.2.1 and" will be changed to "(Actions L.2.1 and L.2.2)" in the fourth line of the second paragraph of the text of Bases, Actions L.1, L.2.1 and L.2.2 on page 16B.3.3-77 of PSAR as a result of the response to the question stated above.

5.

(1) At Lungmen, control rod scram time requirements are essentially no different from other operating BWR plants. One exception is that the Lungmen values of scram insertion times to the specified scram insertion positions are different since they are plant specific.

Limits are specified as a function of reactor pressure to account for the sensitivity of scram insertion times with pressure and to allow a range of pressures over which scram time testing can be performed.

There are four reed switches located at 10%, 40%, 60%, and 100% rod insertion positions which are specifically designed in ABWR for scram time measurements. The scram reactivity used in DBA and transient analyses is based upon the assumed 100% rod insertion position scram time. Historically, control rod position at notch 13 was selected since the results of DBA and transient analyses showed scram reactivity before passing this position is sufficient to bring the reactor to hot shutdown conditions in response to a scram initiation. Since Lungmen will be equipped with FMCRD, a 60% rod insertion position is specified in the control rod scram time criteria which is compatible to notch position 13 based on the discussions above.

[Lungmen TS is consistent with the ABWR SSAR Chapter 16 Specification 3.1.4 which was approved without specifying the 100% rod insertion position scram time limit requirement. The imposed scram time requirements were derived during earlier phases of the ABWR certification program based on the expected FMCRD mechanical capabilities. Historically, the 100% rod insertion position scram time is not required to be verified in accordance with the TS of other BWR plants. As it was stated above, the results of DBA and transient analyses showed that scram reactivity before passing 60% rod insertion position is sufficient to bring the reactor to hot shutdown conditions.]

[Lungmen NPS does have the capability to measure the 100% rod insertion position scram time in its design. Therefore, for consistency with Table 15.0-5, Table 16.3.1.4-1 in the PSAR will be modified to include the requirement of verifying the 100% rod insertion position scram time limit.]

(2) At Lungmen, "slow " rods are defined in the same way as Kuosheng Technical Specifications, i.e., OPERABLE control rods with scram times that are not within the limits of Table 16.3.1.4-1 are

considered "slow" based on NOTE 1 of PSAR Table 16.3.1.4-1.

When work that could affect the scram insertion time is performed on a control rod, the scram testing must be performed before declaring the control rod operable. SR 3.1.3.4 is performed in conjunction with the control rod scram time testing of SR 3.1.4.1, SR 3.1.4.2, SR 3.1.4.3, and SR 3.1.4.4 to demonstrate that the affected control rod is still within the limits of Table 16.3.1.4-1 for startup conditions. Since 60% rod insertion position scram time limit in Table 16.3.1.4-1 is applicable for reactor pressure < 6.55 MPaG, SR 3.1.3.4 is only required to be performed at this low reactor pressure condition during plant startup.

Additionally, control rods which fail the 60% rod insertion position scram time limit specified in SR 3.1.3.4 are considered "inoperable" and are not "slow" in accordance with Table 16.3.1.4-1 NOTE 2 of PSAR. Measurement of the scram times with the reactor pressure ≥ 6.55 MPaG demonstrates acceptable scram times for the transients analyzed in the PSAR. However, drives which fail to meet a low pressure criterion invariably also fail to meet the high pressure criterion.

[Please note that the requirements of each SR are clarified in the BASES section in PSAR. After the work that could affect the scram insertion time is completed on an "affected" control rod and prior to declaring control rod operable, scram time testing must be done to demonstrate that each affected control rod retains adequate scram performance over the entire range of applicable reactor pressure from zero to the maximum permissible pressure in accordance with SR 3.1.4.3.]

[SR 3.1.4.3 "implicitly" specifies the "test conditions" by requiring each affected control rod scram time within the limits listed in Table 16.3.1.4-1 under the column of different "reactor steam dome pressures", i.e., < 6.55 MPaG for startup condition and reactor steam dome pressure ≥6.55 MPaG.]

[SR 3.1.3.4 is required to be conducted at a frequency in accordance with the scram time testing of SR 3.1.4.3 during low reactor pressure condition. For reactor steam dome pressure < 6.55 MPaG, only 60% rod insertion position scram time testing will be performed for startup conditions since only 60% rod insertion position scram time limit is applicable to the result of the measured scram time for the affected control rod. Therefore, no change will be made to the PSAR as a result of this response.]

No changes will be made to the PSAR as a result of the response to the question stated above.

ROCAEC Review Comments:

Item 3.(2) : Similar to Required Action C/D, please consider adding the following "Place one inoperable channel in trip"; Completion Time will be "Immediately" for Action G so that it is consistent with what is in the Bases.

Item 3.(5) : Please clarify whether the Channel Function Test will cover "Comprehensive Function Test" ? If not, the response is not adequate. Also, why SR 3.3.1.2.4 should be executed when "Reactor Mode Switch shutdown Position" and "Manual MSIV Actuation" ?

Item 4.(2) : DIV II power source lost could cause DIV II/IV failure simultaneously so the actions when any one DIV I/III fails and any one DIV II/IV fails should be different because assuming any one DIV II fails, if DIV II power source is lost, 3 divisions will become inoperable simultaneously and safety functions will be lost. And if any one DIV II/IV fails, coupled with another single failure that makes any one DIV I/III inoperable, then the other 2 divisions can still perform safety functions. So this item needs further explanation.

Item 5 : TPC response did not clearly respond to the question why the Technical Specification did not include the scram time when the control rod is 100% inserted. The 4th line of the second paragraph of PSAR 15.0.4.4.1 stated that "Technical Specification Scram Speed in Table 15.0-6 (Which should be

Table 15.0-5) is used in the Safety Analysis..." and there are 4 positions adopted in Table 15.0-5 (namely, 10%, 40%, 60% and 100%). But Table 16.3.1.4-1 of Technical Specification only listed 3 positions (i.e., 10%, 40% and 60%) and 100% Rod Insertion was not included. Please provide further explanation to this inconsistency.

TPC Review Comments on GE's Further Clarifications:

Item 3.(2) Similar to the base of Action C.2 on page 16B.3.3-74, the base of Action G.1 on page 16B.3.3-75 which described "restore the actuation logic to ½ ..."

According to the bases, we suppose that one affected inoperable channel must be placed in trip. Furthermore when three of four output channels for same function become inoperable, the action should be taken to force the actuation logic to become 1/1. So a protective Action from the function is still available.
Please consider to incorporate ROC-AEC comment for consistency with the bases.

Item 3.(5) We note that the Channel Functional Test will not cover the Comprehensive Functional Test. However, please clarify whether the only performed Channel Functional Test can meet the ITS requirement.

Item 4.(2) GE response to this question is only considering the status of 2 or 3 divisions inoperable. We accept those actions for 2 or 3 divisions inoperable condition. However, we still need your explanation on 1 division inoperable.

There should have different actions for 1 division inoperable condition, because when system operation is under this condition, the system can become 2 or 3 division inoperable.

As ROC-AEC's comment, Division II power lost could cause DIV II/IV failure simultaneously, so the action should be different depending on any one of DIV/III failure or any one of DIV II/IV failure existing in advance.

Assuming any one of DIV I/III fail, if DIV II power source is lost, then 3 Division will become inoperable, and if any one of DIV II/IV fail, coupled with any single failure that takes any one DIV I/III inoperable, then 2 Division becomes inoperable.

Further Clarifications:

Item 3.(2) As stated in the previous response, Action C/D applies when three or all of the LOGIC CHANNELs of the same Function become inoperable, while Action G applies when three or four OUTPUT CHANNELs for the same Function are inoperable. Although the number of inoperable channels are the same, the Completion Time and the Required Actions for OUTPUT CHANNELs were approved by the USNRC and were shown to be more restrictive than that of LOGIC CHANNELs during ABWR certification.

Action G requires to restore at least two output channels to operable status within an hour which is more restrictive than placing one inoperable channel in trip immediately and restoring at least one inoperable channel within six (or one) hours as required by Action C/D. Please note that Action G was approved by the USNRC during ABWR certification and is applicable to an entry condition with either three or four inoperable channels. Within an hour and before at least two output channels are restored to operable status, the output channels are inoperable regardless whether one of the inoperable channel is placed in trip or not. With at least two output channels restored to operable status within an hour, it is not necessary to put one remaining inoperable channel(s) in trip. Therefore, the proposed action to place one inoperable channel in trip immediately is not necessary.

However, in order to further clarify Action G, the following changes will be made to the PSAR as a result of this response to the Additional AEC Question stated above:

On page 16B.3.3-75, third line of the first paragraph of Bases for Action G.1, add "(one channel fails tripped)" after the word "1/1".

On page 16B.3.3-75, second and third lines of the second paragraph of Bases for Action G.1, change the statement "restores the actuation logic to ½ so plant protection is maintained for a single additional failure" to "causes the actuation logic to become 2/2 so some degree of plant protective action is restored."

On page 16B.3.3-75, second line of the third paragraph of Bases for Action G.1, add "on at least two of the inoperable channels" after the word "repairs".

Item 3.(5): The Comprehensive Functional Test is a set of tests to exercise each RPS and MSIV closure functions by simulating accident events that vary the inputs and outputs of the RPS/MSIV actuation logic. However, no specific PSAR safety analysis takes credit for the Manual RPS Scram and Reactor Mode Switch Shutdown Position, and Manual MSIV Actuation function. These functions are provided for overall redundancy and diversity of the reactor scram and isolation function, respectively. Manual divisional control switches for reactor scram and main steamline isolation are independent of microprocessor-controlled logic. The actuation circuitry for these functions are hardwired and their locations near the load drivers for the scram pilot valve and MSIV solenoids.

A Channel Function Test for the manual scram and main steamline isolation functions will meet the ITS requirements since it is intended for plants with hardwired logic and data processing systems. The Comprehensive Functional Test is intended to provide end-to-end testing specifically for most SSLC process variables multiplexed via the EMS. The Channel Function Test will not cover but overlaps the comprehensive function test requirements. The combined or overlapping Channel Functional Test provides complete end-to-end testing of the Comprehensive Functional Test requirements on all RPS and MSIV protective functions.

The comprehensive functional test (SR 3.3.1.2.4) is, therefore, not required to be performed on the Manual MSIV Actuation channels.

Therefore, as a result of this response to the Additional AEC Question, surveillance requirements of Manual MSIV Actuation and Reactor Mode Switch Shutdown Position in Table 16.3.3.1.2-1 will be changed from "SR 3.3.1.2.4" to "SR 3.3.1.2.1" which is consistent with the Manual RPS Scram function.

Item 4.(2): For both LCO 3.3.1.1. and 3.3.1.2, a condition with both Div. II and IV failure as a result of Div. II power failure and any one of Div. I or III failure will direct the operator to the Required Actions associated with entry Condition C.

Additionally, a condition when either Div. II or IV fails due to reasons other than Div. II power failure coupled with any one of Div. I or III failure will direct the operator to the Required Actions associated with entry Condition B. Please refer to Bases and further explanations discussed in Specifications 16B.3.3.1.1 and 3.3.1.2.

A further investigation as summarized below concluded that no specific actions will be required for the case with one division inoperable as described in the question:

Case I: Assuming DIV I fails, then Condition A is entered with the specified Actions. If DIV II power is lost within the period of specified Completion Time for the Required Actions to be taken, then a total of three channels become inoperable. This will direct the entry of Condition C and the implementation of associated Required Actions. If any DIV II, III or IV fails within the specified Completion Time for the Required Actions to be taken, then a total of two channels become inoperable. This will direct the entry of Condition B and the implementation of associated Required Actions.

Case II: Assuming DIV III fails, then Condition A is entered with the specified Actions. If DIV II power is lost within the period of specified Completion Time for the Required Actions to be taken, then a total of three channels become inoperable. This will direct the entry of Condition C and the implementation of associated Required Actions. If any DIV I, II or IV fails within the period of specified Completion Time for the Required Actions to be taken, then a total of two channels become inoperable. This will direct the entry of Condition B and the implementation of associated Required Actions.

Case III: Assuming DIV II fails, then Condition A is entered with the specified Actions. If any DIV I, III or IV fails within the period of specified Completion Time for the Required Actions to be taken, then a total of two channels become inoperable. This will direct the entry of Condition B and the implementation of associated Required Actions.

Case IV. Assuming DIV IV fails, then Condition A is entered with the specified Actions. If any DIV I, II or III fails within the period of specified Completion Time for the Required Actions to be taken, then a total of two channels become inoperable. This will direct the entry of Condition B and the implementation of associated Required Actions.

No change will be made to PSAR as a result of this response to the Additional AEC Question stated above.

Item 5: As it was stated in the previous response, the 3 positions scram time requirements currently presented in Table 16.3.1.4-1 of Lungmen TS are consistent with the ABWR SSAR Chapter 16, Specification 3.1.4. This imposed scram time requirements was derived based upon the expected FMCRD mechanical capabilities during earlier phase of the ABWR certification program.

However, Lungmen design does have the capability to measure 100% rod insertion position scram time. The 100% rod insertion position scram time limit will, therefore, be added to Table 16.3.1.4-1 for consistency with Table 15.0-5 as a result of this response to the Additional AEC Question stated above.

ROCAEC Review Comments:

Item 3.(2):

(1) Please explain why Action G is more restrictive than Action C?

(2) Before the above question is clarified, action to modify Bases should be temporarily suspended.

Item 3.(5):

(1) Please consider adding an item "logic system function test" according to BWR-6 ITS.

(2) GE proposed to change the "Reactor Mode Switch Shutdown Position" surveillance test requirement into SR 3.3.1.2.1 and perform one Channel Function Test every 7 days. Can this test be performed during normal operation? Please clarify.

Further Clarifications:

Item 3.(2).(1)

Entry condition G is more restrictive than entry condition C, since:

a. The Required Action for entry condition G is to repair at least two inoperable output channels to operable status while the Required Action for entry condition C is to repair at least one inoperable logic channel to an operable status and placing one of the other inoperable logic channel in trip condition. Since Action G.1 requires more channels to be repaired than that required by the combination of Actions C.1 and C.2, it is therefore more restrictive.

b. The "total" Completion Time for implementing the Required Action associated with the entry condition G is one hour, while the "total" Completion Time for implementing the Required Actions associated with the entry condition C is six hours. The "total" Completion Time for implementing Action G.1 is shorter than that required by the combination of Actions C.1 and C.2.

Therefore, it can be concluded that Action G.1 is more restrictive since it requires repairs of more channels in a shorter period of time than the combination of Actions C.1 and C.2.

Item 3.(5).(1)

Agree with the comment.

As stated below, the PSAR surveillance requirement for the RPS actuation function on "Reactor Mode Switch Shutdown Position" will be changed from "Channel Functional Test" to "Logic System Functional Test" at a frequency of 18 months.

Item 3.(5).(2)

Further investigation revealed that the proposed change in the previous response had not incorporated the result of introducing an unplanned transient during the plant operation if the Channel Functional Test is performed at the specified frequency. As a result of this, a Logic System Functional Test shall be performed as appropriate on all required contacts and trip functions associated with the Reactor Mode Switch Shutdown Position rather than Channel Functional Test.

Therefore, the following changes will be made to the PSAR as a result of the above response to the ROCAEC Further Comments :

a. On page 16.3.3-20, add "SR 3.3.1.2.8 Perform LOGIC SYSTEM FUNCTIONAL TEST" to the end of listed surveillance requirements under the "SURVEILLANCE" column and add "18 months" to the end of listed surveillance requirements under the "FREQUENCY" column.

b. On page 16.3.3-21, surveillance requirement of Reactor Mode Switch Shutdown Position function in Table 16.3.3.1.2-1, change "SR 3.3.1.2.4" to "SR 3.3.1.2.8".

c. On page 16B.3.3-81, add the below listed statements after the first paragraph, as:

SR 3.3.1.2.8

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific Function. The 18 months frequency is based on the ABWR expected refueling interval and the need to perform this Surveillance under conditions that apply during a plant outage to reduce the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the specified frequency. This provides confidence that the specified frequency.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 16-021

PSAR Sections: 16.3.5

Question Date: June 10, 1998

PSAR Question:

1. Please explain whether the Completion Time of Actions described in this chapter was relaxed because of the analysis in PSAR 6.3.3.9 ? If yes, then since the analysis will be performed in FSAR only, isn't it more appropriate to describe the Completion Time as interim values (denote it by [ ] symbol) ? If not, then please explain the reason why it is relaxed.
2. What is the bases for the Completion Time of once every 8 hours stipulated in Required Action B.1.2 to verify the swing DG circuit breakers are capable of being aligned to each of the ESF buses ? Is it true that every verification will cause DG power source temporarily unavailable to the ESF BUS ? If yes, then is this action and its frequency going to adversely affect safety ? If it helps safety, then why it was not included in Required Action of Condition C ?
3. SR 3.5.1.5 stated that "develop a flow $\geq$ 182 m^3/h against a system head corresponding to reactor pressure" but the upper limit of the RCIC discharge head is 8.12 MPaG. So, should the reactor pressure be changed to $\geq$ 8.12 MPaG to be in line with HPCF ?
4. The descriptions in SR 3.5.1.7 and SR 3.5.1.8 are identical. Please correct it.
5. Why the verification of CST level was not included in the Surveillance Requirements in 16.3.5.1 (like SR 3.5.2.2) ?
6. The Completion Time was said to be 1 day in the Bases for Action B.2 (page 16B.3.5-6) which is different from this section. Please explain and make necessary corrections.

Response:

1. Core cooling LOCA analyses covering the complete spectrum of postulated breaks were performed with only one RHR system in the LPFL mode and 5 ADS valves available. These calculations are also bounding for the case with one HPCF subsystem and 5 ADS valves

available since, compared to the RHR/LPFL, the HPCF subsystem has the additional capability to inject at high reactor system pressures.

The relaxation was based on the Lungmen ECCS design which is expected to result in improved ECCS performance since there is a considerable margin between the number of installed ECCS subsystems and the number of ECCS subsystems used for LOCA analyses. Originally, GE proposed a 30 days ECCS LCO completion time during the ABWR certification process. However, the USNRC has reviewed and approved the 14 days completion time which was adopted in the Lungmen Technical Specifications.

Although LOCA analyses for Lungmen is still in progress, the ECCS alignment assumptions shall remain the same, i.e., only one RHR system in the LPFL mode and 5 ADS valves available. The results shall also remain the same which are expected to meet the ECCS acceptance criteria set forth in 10CFR50.46. However, the "Completion Time" values, expected to be established for each ACTIONs in Specification 16.3.5, should be bracketed (i.e., [xxxx]) until they are further supported by the core cooling LOCA analysis,which is to be supplied with the FSAR submittal.

Therefore, each proposed "Completion Time" value in Specification 16.3.5 of PSAR will be bracketed, i.e., [xxxx], as a result of this response to the comment of question stated above.

2. When RCIC or RCIC in combination with any one other ECCS subsystem becomes inoperable, it is necessary to verify proper circuit continuity initially and at a more frequent basis thereafter, i.e., once per every 8 hours, to ensure a highly reliable power source remains for swing DG electrical power supply to one of the three ESF buses distribution network. Please note that during the course of verifying the required action, each ESF electrical supply distribution network circuit remains in its normal lineup and will not become inoperable. Therefore, this verification does not impose adverse effect on plant safety.

Required Actions C.1 will ensure a functional ACIWA mode of RHR C subsystem to provide core cooling during a station blackout. In addition, Condition B is entered after restoring one of the two inoperable ECCS subsystem per Action C.2. This action will ensure adequate core cooling during a LOCA is provided in addition to at least one HPCF subsystem. As such, it is not necessary to duplicate the Required Action B.1.2 for Condition C under LCO 16.3.5.1 in the PSAR..

No changes will be made in PSAR as a result of the response to the question stated above.

3. Since RCIC pump is turbine driven, required steam dome pressure must be available to perform the RCIC pump flow tests. SR 3.5.1.5 is performed to trend the system performance and ensure that system capability to provide rated flow is not degraded at higher operating ranges in accordance with the IST program. SR 3.5.1.6 is performed to verify the operability of the RCIC at lower operating range after plant recovery to operation from refueling outage.

   During flow testing, the RCIC pump flow rate is typically confirmed at a pump discharge pressure which is adequate to overcome losses due to elevations between the RCIC pump suction and the discharge to the reactor, the piping friction, and the primary system pressure, i.e., the reactor pressure at which the surveillance testing SR 3.5.1.5 and SR 3.5.1.6 are to be performed. As such, SR 3.5.1.5 and SR 3.5.1.6 are consistent in requiring the verification of RCIC pump's capability to develop a rated flow against a system head corresponding to the "operating" reactor pressure. The changes that are recommended in the question stated above are, therefore, not applicable to the RCIC system.

   No changes will be made in the PSAR as a result of the response to the question stated above.

4. Although the descriptions in SR 3.5.1.7 and SR 3.5.1.8 are identical, the notes associated with these two SRs specifically require that vessel injection and valve actuation are to be verified for surveillance tests SR 3.5.1.7 and SR 3.5.1.8, respectively. Please note that as stated in the

BASES of surveillance requirements, SR 3.5.1.7 covers ECCS subsystems such as HPCF, RCIC and RHR/LPFL, while SR 3.5.1.8 addresses ADS designated S/RVs. For consistency, however, the descriptions in SR 3.5.1.7 and SR 3.5.1.8 in PSAR will be modified.

Therefore, the following changes will be made to the listed PSAR sections as a result of the response to the question stated above, i.e., add "except ADS" after "ECCS subsystem" in SR 3.5.1.7 and change "each ECCS" to "ADS" in SR 3.5.1.8 on page 16.3.5-5 .

5. Specification 16.3.5.1 does not require the verification of CST level in surveillance requirement since water source in the CST was not credited in LOCA analyses. Please note that the verification of suppression pool water level is required on a more frequent basis in SR 3.6.2.2.1 in modes 1, 2, and 3 to ensure the availability of adequate suppression pool water source for ECCS pumps.

   Verification of CST water level in mode 4 is required in SR 3.5.2.2to incorporate the shutdown safety issue imposed by the USNRC during ABWR certification. In addition, there is an "or" between two parameters specified in SR 3.5.2.2, i.e., suppression pool water level and CST water level. Therefore, either one of the water sources, i.e., suppression pool or CST, above the specified minimum water level will ensure adequate NPSH is provided for HPCF pumps while the plant is in mode 4 operation.

   No changes will be made in PSAR as a result of the response to the question stated above.

6. This is an editorial error.
   For consistency, in the 10th and 11th lines from the last paragraph of page 16B.3.5-6, change "1 day" to "7 days" for the completion time stated in the BASES for Action B.2 as a result of the response to the question stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

ROCAEC Review Comment:

Item 2 : Please explain how to verify that the swing DG can be connected to each ESF bus? Is it necessary to perform breaker switch operation ?

Further Clarification:

Item 2: Capability of swing DG to connect to each ESF bus can be verified by visual observations of correct swing DG circuit breaker alignments. This is similiar to the verification of circuit breakers lineup for other systems during plant operation. It is not necessary to perform breaker switch operation during this verification.

Using 16.3.5.1, ECCS -Operating, Required Actions B.1.1 and B.1.2, the surveillance requirements are as follows:

B.1.1 Perform swing DG surveillance testing in accordance with the applicable surveillance requirements if not performed within seven days and every seven days thereafter. This surveillance would include breaker switch operation performed to provide a positive functionality test. This test would be accomplished in the following manner: first, place the Divisional DG under test in the "MAINTENANCE" mode (switch located on the local control panel) in order to prevent it from responding to emergency signals; and secondly, manually align the swing DG to the Division under test by operating the appropriate breakers. Once aligned, the swing DG replaces the Divisional DG and will respond to any signal received. Upon completion of the testing, the tie breakers are opened, the Divisional DG returned to "REMOTE" mode and the swing diesel is returned to normal standby. (See also 16B.3.5.1 ECCS-Operating Bases, Actions A.1, B.1.1, B.1.2, B.2, and B.3 page 16B.3.5-6).

B.1.2 Perform visual inspection of swing DG circuit breaker alignment capability every 8 hours thereafter.

No changes will be made to PSAR as a result of this response to the Additional AEC Question stated above.