編　輯　說　明

一、本問題與答覆彙編係原能會專案小組審查台電公司龍門計畫初期安全分析報告書，所提出之問題以及台電公司針對各項問題所做之答覆與說明，爲原能會審查結論報告參考附件之一。

二、原能會在審查作業期間，共向台電公司提出五百九十八件審查意見，各章審查意見之統計，請參見表一。

三、由於資料繁多，爲利於參閱，本彙編計分四冊付印，各冊涵蓋章別，請參見表二。

四、各冊內容均概分爲「審查意見摘要表」及「審查問題與答覆內容」兩大部份，謹供對照查閱。

表一　各章審查意見統計表

章別	章名	審查意見數
一	電廠簡介	26
二	廠址特性	18
三	結構、系統、組件與設備之設計	63
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五	反應器冷卻水系統及其相關連系統	21
六	特殊安全設施	24
七	儀控系統	51
八	電力系統	8
九	輔助系統	42
十	蒸汽與動力轉換系統	15
十一	放射性廢料管理	30
十二	輻射防護	47
十三	運轉管理	3
十四	初始試驗計畫	11
十五	事故分析	38
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附錄 A	安全度評估	21
附錄 B	整體可靠度	2
附錄 C	緊急計畫	52
附錄 D	後端營運計畫	6
附錄 E	經驗回饋	3
合計		598

表二　各冊涵蓋章別表

冊別	涵蓋章別
四之一	第一章、第二章、第三章、第四章、第五章
四之二	第六章、第七章、第八章、第九章
四之三	第十章、第十一章、第十二章、第十三章、第十四章、第十五章、第十六章
四之四	第十七章、第十八章、第十九章、附錄 A、附錄 B、附錄 C、附錄 D、附錄 E

審 查 意 見 摘 要 表

（ 第一章 ～ 第五章 ）

第一章　電廠簡介

編號	內　　　容
01-01	Complex plot plan 圖不夠清晰。
01-02	比較 ABWR 和核四廠熱效率和廠內負載。
01-03	請提供 ABWR 和 K-6/7 設計比較。
01-04	澄清「電廠潛在放射性排放均予以監測」敘述。
01-05	1.何時提出 life cycle 管理方案及老化管理計劃？如何與設計過程配合？
	2.台電引用 EPRI-URD 及 NPAR 方案要求之立場。
01-06	RPV 是否執行 100%銲道檢查？能執行 ID 和 ODUT 檢測？
01-07	澄清 RHR 系統設計壓力？
01-08	澄清寒水系統和廠房冷卻可用性。
01-09	10 CFR 50.65 執行方案何時提出？
01-10	ISI 所適用之 ASME BPV 規範年版。
01-11	1.請舉例說明被動組件故障。
	2.SLC 系統管路及槽是否考量單一故障？
01-12	表 1.8-22 列舉了許多將納入設計之經驗，請澄清是否已納入。
01-13	1.控制室適居區是否評估 DBA 時，可能影響之射源或毒氣源？
	2.系統中不凝結氣體之排放是否有獨立的處理系統？
01-14	當 LOCA 時 RBCW 之隔離閥會關閉，其對 RMC 之影響如何？
01-15	1.為何採用雙層熱套管設計而不採用單層熱套管？
	2.雙曾熱套管是否可更換式？與美國某些廠使用之三層熱套管之比較如何？其 ISI 限制為何？
01-16	請補充說明正常運轉時發生 LOCA 而又喪失 RMISS 之情形。
01-17	龍門核電廠所使用之 316 及 316L，其含碳量為 0.08%大於 0.035%甚多。
01-18	表 1.8-中部份法規指引不適用於核四理由為何？
01-19	請比較核一、二及龍門電廠 SBO 之導因及機率。

編號	內　　　　容
01-20	SBO 承受期限和 SEDG 可用時限。
01-21	請比較核四與日本 K-6/7USABWR 之差異及其對安全和運轉、維護上的影響。
01-22	1.請說明 COPING 分析之內涵。
	2.請說明 SBO 時抑壓池設計需符合之設計基礎為何?
01-23	請說明 HPCF 及 RCIC 是否裝設充水系統
01-24	請說明核四廠不需執行低壓力之 SRV 測試之考量。
01-25	高輻射是否會引起 MSIV 關閉。
01-26	台電執行 10CFR 50.65 是否自願性

第二章　廠址特性

編號	內　　　　　容
02-01	澄清廠址區域性地質特性
02-02	補充說明廠址地表斷層及地下水
02-03	討論廠址歷年洪水資料
02-04	討論鄰近區人口分佈統計之預估
02-05	討論有關廠址淡水供應設施及水權
02-06	廠區上壤液化潛在性之澄清
02-07	廠址區域與廠區氣象資料之討論
02-08	廠區地下水狀況及除水系統
02-09	廠區地下層(履蓋層/岩石)狀況改善技術
02-10	廠區氣象資料是否能涵蓋排放煙囪之設計
02-11	廠區地下水是否有湧泉現象
02-12	廠區地質與地下物質特性之澄清
02-13	廠址地下水基座除水系統及地下水特性
02-14	廠區開挖回填之監測計畫及管制稽查作業
02-15	討論廠區岩石機械特性
02-16	廠址古河道之變遷情況
02-17	廠區地基開挖地質剖面圖探討
02-18	核四龍門計畫由 PWR 換為 ABWR 對地基邊坡等之影響

第三章　結構、系統、組件與設備之設計

編號	內　　　　容
03-001	澄清各廠房動態分析。
03-002	澄清系統與次系統間相互關係分析方法。
03-003	OBE Exceedance 不符合 R.G.1.166
03-004	PSAR 中未對 TUNNEL 設計作詳細說明。
03-005	PSAR 中對次圍阻體與隔間屏障的說明不詳細。
03-006	澄清土壤與結構互制分析方法。
03-007	廠房設計中所用程式資料說明不足。
03-008	URS 低壓管路設計壓力之澄清。
03-009	CRD 管路設計壓力之澄清。
03-010	高能量管路與低能量管路定義澄清。
03-011	地震二級管路進行動態分析者是否屬 essential system。
03-012	蒸汽通道中不屬 essential system 者是否要考慮 HELB 之動態效應。
03-013	澄清 class 2 管路之設計準則。
03-014	對 PSAR 中使用 NB-3653 之澄清。
03-015	對地震二級 SSC 再分類在 FSAR 階段完成
03-016	汽渦輪機廠房是否要考慮動態分析？由 seismic guide 到 stop valve 間管路的安全分類澄清。
03-017	煙囪設計的澄清與非地震一級的結構為何不考慮颱風之負載。
03-18	飛機災害分析中機率值澄清。
03-19	地震二級結構損壞之影響準則澄清
03-020	有關地震設計參數之澄清。
03-021	地震分析模式與方法之澄清
03-022	各種負載之澄清
03-023	澄清 Appendix S 之問題。
03-024	非對稱問題之分析與 100-40-40 方法之澄清。
03-025	SRV 的 load factor 澄清。
03-026	圍阻體設計之澄清。
03-027	圍阻體結構混凝土、襯版、反應器支撐設計之澄清。

編號	內　　容
03-028	其他地震一級結構之設計問題澄清
03-029	Waterproofing membrane 對剪力傳遞之影響。
03-030	PSAR 附錄 3D 中計算機程式之說明。
03-031	土壤參數之提供
03-032	地震頻譜問題澄清
03-033	SRP 中管路設計公式之澄清
03-034	地震二級 SSC 之在分類問題
03-035	次圍阻體的 leak-tight barrier 問題
03-036	RCPB 相關之圖示澄清。
03-037	爐心側版經驗回饋。
03-038	風力負載之澄清。
03-039	拋射物問題澄清。
03-040	動態測試問題澄清。
03-041	我國建築法規地震設計與地質問題澄清。
03-042	建議使用多組地震數據
03-043	Sensing line 與 sample line 引用之法規與規範
03-044	URS 設計問題
03-045	飛機災害機率問題之澄清
03-046	有關地震設計參數之澄清。
03-047	地震分析模式問題澄清
03-048	地震分析中所用技術與程式之澄清
03-049	圍阻體模式之澄清
03-050	有關 TANK 問題之澄清。
03-051	廠房結構與地基間負載之傳遞澄清
03-052	表 3.2-3 安全分類之澄清。
03-053	圖 6.2.3.4.1 是否納入相關章節
03-054	針對重要設備組件訂定設計完成之時程
03-055	PIPE WHIP RESTRAINT 分析之澄清
03-056	Class 1 管路斷管分析之使用公式
03-057	Class 1 管路斷管分析之使用公式
03-058	破管分析之準則
03-059	低能量管路破裂準則
03-060	RPV internal 設計負載組合澄清

編號	內　　　　　容
03-061	Safety service pump house 所用規範之說明不足
03-062	依據 SRP 中各廠房中所用材料資料之提供
03-063	汽渦輪機廠房設計之規範、負載組合、分析方法之說明

第四章　反應器

編號	內　　　　容
04-01	GE12 冷爐中子分析之準確度
04-02	停機餘裕度
04-03	長週期冷爐臨界分析之準確度
04-04	反應器分析程式用途說明
04-05	停機餘裕度
04-06(1)	功率/流量圖說明事項澄清
04-06(2)	功率/流量圖 max. rod line
04-07	功率分佈計算參改文件澄清
04-08	控制棒材料選擇
04-09	自動功率調節之功率傾斜
04-10	停機餘裕度
04-11	功率/流量圖之 SCRRI 動作區
04-12	功率/流量圖標示澄清
04-13(1)	核四與 ABWR 燃料束穩定性比較
04-13(2)	核四與 ABWR 爐心穩定性比較
04-14(1)	冷加工奧斯田鐵之規格
04-14(2)	CRD 各組件之材料規格
04-14(3)	Peripheral fuel support 之材料
04-14(4)	Delta ferrite 之含量要求
04-15	CRD 脫接偵測器之失效模式
04-16(1)	核四負載追隨能力
04-16(2)	核四負載追隨範圍
04-16(3)	負載追隨功能在暫態下的行為
04-17	蒸汽分離線之相關問題
04-18	有關 SSE 下之燃料設計基準
04-19(1)	控制棒構造圖
04-19(2)	Lower transition piece 構造
04-19(3)	CRD 使用經驗與失效歷史
04-20(1)	GE12 與 GE 9B 抗震性比較
04-20(2)	GE 的燃料地震設計程序

編號	內　　　容
04-21(1)	17-4PH 張力強度過低
04-21(2)	控制棒 Drive shaft hardening
04-21(3)	控制棒驅動機構之 separation magnet 材質
04-22(1)	不銹鋼之碳含量說明
04-22(2)	焊接材料之 delta ferrite content
04-22(3)	Alloy X-750 之 annealing temperature
04-22(4)	使用 alloy stellite 6 做為 HPCF couplings 之表面硬化材質之說明
04-23	Alloy 600M 相關問題
04-24	提供 FMCRD 測試等資料
04-25(1)	冷加工奧斯田鐵不銹鋼
04-25(2)	爐內組體焊接準則
04-26	HCU 急停能力
04-27(1)	功率/流量圖標示澄清
04-27(2)	功率/流量圖標示澄清
04-28	GE 燃料設計準則
04-29(1)	在 min. flow 下之穩定性
04-29(2)	在 min. flow 下之穩定性
04-29(3)	低自然對流對 RHR 之影響
04-30	GE12 之使用經驗
04-31(1)	附錄 4A 之棒位問題
04-31(2)	附錄 4A 之棒位問題
04-32(1)	SCRRI 工作範圍
04-32(2)	一台 RIP 是否會進入不穩定區
04-32(3)	9 台 RIP 運轉時的 SCRRI 邏輯
04-32(4)	9 台 RIP 運轉時的 SCRRI 邏輯
04-33(1)	9 台 RIP 運轉時之流量分析
04-33(2)	RIP trip 故的 flow reversal
04-33(3)	9 台 RIP 運轉之 CPR 計算

第五章　反應器冷卻水系統及其相關連系統

編號	內　　　容
05-001	Code case N-580 之使用澄清。
05-002	SRV solenoid valves 問題澄清。
05-003	RHR 系統停機模式之溫度要求問題澄清。
05-004	ASME CODE EXEMPTION 問題澄清。
05-005	SRV discharge line 檢測分類問題。
05-006	核四廠水質與材質問題。
05-007	ISI 權責與審查單位之訂定問題。
05-008	反應器壓力槽溫度壓力曲線問題。
05-009	RHR 系統 NPSH 問題。
05-010	ACIWA 事故之 sequence of event 問題。
05-011	淨化水質之 F/D precoat 均勻度問題
05-012	LPFL 的 Q_{min} 問題。
05-013	因應設計基礎事故時的移熱需求。
05-014	SRV discharge 所引發動態負載是否納入主蒸汽管路與飼水管路之設計負載中。
05-015	壓力槽材質之鎳含量與領先因子問題。
05-016	表 5.2-4 之 design exhaust pressure 問題。
05-017	RCIC 的 capacity 問題。
05-018	壓力槽溫度差問題。
05-019	核四廠是否要執行加氫水化學問題。
05-020	有關 Barometric condenser 問題澄清。
05-021	主蒸汽管路因高速流產生跳機問題。

審查問題與答覆內容

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-01-001

PSAR Sections: Figure 1.2-1, Table 1.3-1

Question Date: November 17, 1997

PSAR Question:

1. Figure 1.2-1 not clear, please provide a clear figure.

2. Table 1.3-1, Please include Peaking factors, Void coefficient and Doppler coefficient.

PSAR Response:

1. A clear Figure 1.2-1 is provided as attached.

2. Table 1.3- 1 will be modified to include the comparisons of Design Peaking Factors, Void coefficient and Doppler coefficient. Peaking factors for the Lungmen NPS are provided through the cycle based on projected control rod patterns and are presented in PSAR Figures 4A-4a to 4A-13a.

The void and Doppler coefficients for Lungmen initial core will be provided in the FSAR. Table 1.3-1 will be modified as follows:

Table 1.3-1 Comparison of Nuclear Steam Supply System

	Lungmen NPS	ABWR SSAR	NMP-2 BWR/5	Grand Gulf BWR/6
Design	278-872	278-872	251-764	251-800
Thermal and Hydraulic (Section 4.4)				
Design Power Peaking Factor	*			
Maximum relative assembly power		1.40	1.40	1.40
Local Peaking factor		1.25	1.24	1.13
Axial peaking factor		1.40	1.40	1.40

*Peaking factors for the Lungmen NPS are provided in figures 4A-4a to 4A-13a

Table 1.3-1 Comparison of Nuclear Steam Supply System (Continued)

	Lungmen NPS	ABWR SSAR	NMP-2 BWR/5	Grand Gulf BWR/6
Design	278-872	278-872	251-764	251-800

Nuclear (first core) (Section 4.3)				
Dynamic void coefficient (c/%) at core average voids (%) (EOC-rated output)	FSAR	-5.20 @102% rated output 39.2	-8.57 40.54	-7.14 41.31
Fuel temperature doppler coefficient (c/°C) (EOC-rated output)	FSAR	-0.360	-0.419	-0.396

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-01-002

PSAR Sections: 1.1.10

Question Date: December 2, 1997

PSAR Question:

The gross electrical power output for Lungmen plant is approximately 1371 Mwe (Ref ch1 section 1.1.10), but the gross electrical power output for ABWR is approximately 1356 Mwe. On the other hand, the rated thermal power and the net electrical power output for both designs are the same. Does this imply that the thermal performance is better but the house load is larger in Lungmen plant? Please identify the factors which contribute to the difference observed in both the thermal performance and the house load respectively.

PSAR Response:

The turbine-generator design and capability are defined in detail in the SSAR in Chapter 10 (Steam and Power Conversion). SSAR Figure 10.1-2 (Reference Heat Balance for Guaranteed Reactor Rating) shows that turbine-generator gross electrical output for the ABWR plant is 1,371MWe. The value of 1,356 MWe shown in SSAR subsection 1.1.10 is an approximate value, as the text indicates.

In PSAR Figure 10.1-2 (Heat Balance Diagram-Design Flow/1371MWe) the turbine-generator gross electrical output is also shown as 1,371MWe. Thus for both the ABWR and Lungmen NPS the gross electrical output is indeed 1,371 MWe, and the reason for the difference in subsection 1.1.10 is that the PSAR has been updated to show a more exact value.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 1-01-003

PSAR Sections: Chapter 1

Question Date: December 2, 1997

PSAR Question:

GE summarized differences between the U.S. ABWR design and the K-6/7 project in a letter to the NRC staff dated February 20, 1992. GE updated the list of differences in SSAR Amendment 31. Differences between the K-6/7 design and the U.S. ABWR design are identified and maintained in a controlled list (Design Action List, DAL) for future action. Please provide the above three documents.

PSAR Response:

GE February 20, 1992 letter which submitted a summary of the differences between the U.S. ABWR design and the K-6/7 project to the NRC is attached. Also attached is Appendix 1B of Amendment 31 of the SSAR, submitted to the NRC on July 28. 1993, providing an updated listing of the differences. (Please note that this discussion of differences between the K-6/7 design and that of the ABWR was considered by the Utility owner of K-6/7 to be sensitive material. Therefore, in a subsequent revision to the SSAR, this list of differences was deleted with the concurrence of the NRC. It would be very much appreciated if the ROC-AEC could kindly handle this material in a manner which reflects the sensitivity expressed by the Utility owner of K-6/7.)

With regard to the Design Action List (DAL), it was developed by GE to internally authorize and process design changes from K-6/7 to the U. S. ABWR (e.g. the differences listed in Appendix 1B of Amendment 31). Thus, the DAL is an internal, GE Company Private document. It was not submitted to the NRC and is not available for transmittal to Taiwan. However, with the attached transmittal of GE February 20, 1992 letter and Appendix 1B of Amendment 31, documentation of the major differences between K-6/7 and the U.S. ABWR is being provided to the ROC-AEC.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-01-004

PSAR Sections: 1.2.2.4.1

Question Date: December 2, 1997

PSAR Question:

In Chapter 1 Sect. 1.2.2.4.1 of ABWR SSAR, the following statement is given:
"All effluents from the plant which are potentially radioactive are monitored". This statement has been deleted in PSAR of Lungmen Plant. Please clarify whether this statement is still valid for Lungmen Plant or not.

PSAR Response:

The statement made in the ABWR SSAR that "All effluents from the plant which are potentially radioactive are monitored" is still valid for the Lungmen NPS as well. The Lungmen NPS PSAR makes the commitment to monitor all radioactive release paths within the plant in Section 11.5. As this commitment is provided in Section 11.5 of the Lungmen PSAR, it was deleted from subsection 1.2.2.4.1 as part of the effort to simplify the PSAR Introduction, Chapter 1.

No revision to the PSAR is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number): N-01-005

問題章節(PSAR Section): 1.2.1.3

初提日期(Question Date): 86.11.18

問題內容(PSAR Question):

何時會提出Life Cycle Management Program? 何時會提出Aging Management Plan? 如何與design process配合？

問題答覆(Response):

台電Bid Spec. Chapter 1,Section 11.3與Section 8.2.6及GE'S Proposal Appendix A Chapter 1 Section 11.3與Section 8.2.6規定GE公司必須在電廠任何永久設備運轉前的2年及1年分別完成建立電廠設備預防維護檢查計畫建議書(包括設計壽命分類系統及狀態的監視）以及電廠環境系統監視建議書。另1998年4月6日GETP-1998-0366 GE回電通知將老化管理計畫包括在IRA(Integrated Reliability Analysis)計畫內列入FSAR中一起發行供台電使用。

為了符合台電龍門計畫招標規範，核能電廠設計壽命至少40年的要求以及保證日後電廠安全運轉的功能，GE公司允諾在電廠設計過程中由最初的設計階段起就貫徹執行壽命周期管理計畫(Life Cycle Management Program)進行設計，包括設計壽命、狀態監視、暫態界定、儀器及控制的設計、環境及監視、電廠資料庫(Plant Data Documentation)之設計理念，以及20年換照如除污設計等。同時遵循中華民國核能法規以達成設備可接近性（Accessibility），材料選擇最佳化（Optimization），水泥牆防止污染(Prevention Of Concrete Contamination)以及除汙概念計畫(Conceptual Decommissioning Plan)。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編 號(Track Number): N-01-005

問題章節(PSAR Section) 1.2.1.3

初提日期(Question Date): 86.11.18

問題內容(PSAR Question):

台電對引用EPRI—URD及NPAR方案之要求之立場為何？

問題答覆(Response):

對於引用EPRI—URD之設備設計壽命要求及USNRC—NPAR核能電廠老化研究分析報告的目的，是為了強調符合龍門計畫ALWR核能電廠設計壽命至少40年之規範要求。台電公司(包括GE公司）遵照規定允諾在設計之初就慎重地建立設計壽命計畫，亦即壽命周期管理計畫(Life Cycle Management Program)。作為電廠由設計階段的設計理念到運轉時的安全功能監視不可或缺的一部分，以提供龍門ALWR成熟機組40年設計壽命多一層的保障。故台電公司對GE引用EPRI--URD及NPAR方案持肯定立場。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-01-006

PSAR Sections: Ch 1.2.2.6.12

Question Date: November 18, 1997

PSAR Question:

Inservice Inspection Equipment :

This section mentioned that GERIS-2000 will be used for RPV inside and outside diameter inspection and SMART-2000 will be used for piping inspection. Please clarify :
Does the Lungmen RPV design allow UT inspection be performed for RPV ID and OD ? and also 100% welding inspection (PSI and ISI) is possible according to 1989 ASME Code Sect XI ?

Response:

The RPV seam welds can be inspected from both the ID and the OD using the GERIS-2000 Data Acquisition System in conjunction with existing GE automatic scanning devices. The RPV nozzle-to-shell welds will be 100% accessible for preservice inspection using a combination of automated and manual UT scanning techniques, but may have limited areas that will not be accessible from the outer surface for inservice examination.

Essentially 100% of the pressure retaining shell welds can be volumetrically examined using UT techniques per the requirements of the 1989 Edition of the ASME Boiler and Pressure Vessel Code, Section XI. Table IWB-2500-1. In a few local areas the access for ultrasonic transducers is less than 2.5T + 50 mm away from the weld seam on one side.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-01-008

PSAR Sections: Ch 1

Question Date: December 9, 1997

PSAR Question:

Please clarify in the Lungmen PSAR whether the two safety issues have been resolved : Leakage Through Electrical Isolators in Instrumentation Circuits as listed in 1C. 2. 53 Issue 142 and Availability of Chilled Water Systems and Room Cooling as listed in 1C. 2. 54.

Response:

1C.2.53 Issue 142

As recognized by the U. S. NRC in NUREG 1503, Safety Issue 142 is considered resolved for the ABWR. Accordingly, the Lungmen NPS PSAR will be revised to add the following paragraph at the end of Section 1C.2.53:

"This issue is considered resolved for the Lungmen NPS with the commitment of TPC to perform the following actions:

1. Annual inspection and testing of all electrical isolation devices between Class 1E and non-Class 1E systems.

2. Repair or replacement of isolators that fail the tests."

1C.2.54 Issue 143

As recognized by the U. S. NRC in NUREG 1503, Safety Issue 143 is considered resolved for the ABWR. Accordingly, the Lungmen NPS PSAR will be revised to add the following paragraph at the end of Section 1C.2.54:

"This issue is considered resolved for the Lungmen NPS based on the above described features and commitments."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-01-009

PSAR Sections: Ch 1

Question Date: December 9, 1997

PSAR Question:

In 1C. 2. 55 Issue 145 it was mentioned that "A program that complies with 10CFR50. 65 will be developed and implemented." Please clarify when this program will be developed and implemented.

PSAR Response:

Taiwan Power Company will develop maintenance rule program one year before Unit 1 fuel loading. The program will be developed based on the information that will be contained in Lungmen FSAR Appendix A PRA, Appendix B IRAP analysis results, and appropriate plant structure and equipment information that is expected to become available during pre-operational testing stage. The component performance data that is obtained during pre-operational testing will provide input for condition oriented goal setting for Structure, System and Components (SSCs).

The maintenance rule program will be evaluated and updated during Unit 1 power ascension test program and first cycle operation. Maintenance rule will be implemented for Lungmen after completion of Unit 1 first refueling outage.

Maintenance rule program will become applicable to Unit 2 when the unit begins pre-operational testing phase. Unit 2 will implement maintenance rule at the same time when the rule becomes effective for Unit 1.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-01-010

PSAR Sections: Ch 1

Question Date: December 9, 1997

PSAR Question:

In Table 1.8-21, please clarify the edition or principle of ASME B&PV code Sec XI (Inservice Inspection).

Response:

Per the US Code of Federal Regulation, the editions and addenda to ASME B &PV Code, Section XI should be applied as follows:

With reference to, 10CFR50.55a (g)(3)(i), the ASME Boiler and Pressure Vessel Code edition including addenda used as construction code shall be applied for the pre-service examination (PSI) of Class 1, 2 and 3 components and their supports. For Lungmen, the code edition to be used for PSI would be the 1989 Edition. Furthermore, 10CFR50.55a (g)(4)(i) requires that ISI during the initial 120-month inspection interval must comply with the requirements of the latest edition and addenda of the code in effect on the date 12 months prior to the date of issuance of the operating license. ISI during the successive 120 months inspection intervals must comply with the latest edition and addenda of the code incorporated 12 months prior to the start of the 120 month inspection interval as outlined in 10CFR50.55a (g)(4)(ii).

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 01-007

PSAR Sections: 1C.2.45

Question Date: December 16, 1997

PSAR Question:

In the subsection, the statement said that "the RHR System can operate at a pressure up to about 3.5 Mpa. Isolation valves (at least two) and piping to the primary system are designed for about 8.6 Mpa." But in the SSAR, the pressures are 3.53 MPa and 8.73 Mpa respectively. Please explain the differences.

PSAR Response:

The value 3.53 MPa was changed to 3.5 MPa in the PSAR because of a round off to achieve consistency in the use of significant figures within this section. The value 8.73 Mpa in the SSAR was inadvertently incorrect since this value represented absolute pressure. The value was changed to 8.6 Mpa in the PSAR, which is the correct gauge pressure value.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-01-0011

PSAR Sections: Ch 1

Question Date: November 19, 1997

PSAR Question:

In Section 1.2.1.1.2 Safety design criteria (11) it was stated that : "safety-related actions are provided by equipment of sufficient redundance and independence so that no single failure of active components, or of passive components in certain cases in the long term, will prevent the required actions." Please explain the following :

1. describe "passive components failure" with examples.
2. Standby liquid control system will be performing safety-related actions. Has consideration been given to single failure of passive components to the piping and tank of this system ? If not, what is the reason ?

PSAR Responses:

Response to Part 1:
A passive component failure is the blockage of a process flow path or failure of a component to maintain its structural integrity or stability, such that it cannot provide its intended nuclear safety function upon demand. Examples would be long term piping failure, pump or valve seal deterioration and mechanical failure of a valve impeding flow.

Response to Part 2:
The Standby Liquid Control System completes its function in the short term. Thus, by the definition provided in Criterion (11), the single failure of passive components does not apply to the piping system and tank of the Standby Liquid Control System.

No PSAR revision is proposed in response to this question.

ROC-AEC Review Comments:

1. It takes about 2.5 hours to inject boron into the core according to PSAR page 9.3-13 Standby Liquid Control System. If this is called short term, then how long will it be called long term?

2. Is it possible to provide more realistic examples? Which pipe of which safety system has been designed with passive failure assumptions? Are these design considerations exceptions or common to all?

Further Clarification:

1. Page 9.3-13 does indeed specify that the boron injection time will be no longer than two and one half hours for the lowest boron injection rate of 8 ppm/min. However, the expected boron

injection rate is 20 ppm/min. which yields boron injection time of about one hour for sufficient boron to be injected to reach cold shutdown. Actually, the reactor will reach hot shutdown in approximately 15 minutes at the expected boron injection rate of 20 ppm/min. Times equal or less than these durations are considered short term for a passive component, subject to periodic surveillance, maintaining its integrity. See item 2 below for an example of consideration of a passive failure in the long term.

2. The RHR piping has been designed with consideration of passive failures. It is because the RHR system may be required to operate post LOCA for up to 100 days that it has been designed consistent with the possibility of a long term passive failure. To accommodate the potential of a long term passive failure, the RHR system has been divided into three loops. When a passive failure is postulated in one of the loops, the remaining loops are able to satisfy the required functions.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-01-012

PSAR Sections: Ch 1

Question Date: November 9, 1997

PSAR Question:

In Section 1.8.3 it was stated that "Table 1.8-22 lists the experience information that has been included in the design". However, lot of the experience information in GESSAR belongs to COL(Combined Operating License). Has those COL related experience information been included in the design or is it just a commitment that it will be included in the design ? If it is only commitment, then please modify the Table to show the actual condition.

Response:

The Generic Letters, IE Bulletins, IE Information Notices, IE Circulars and NUREGs listed in Table 1.8-22 have all been reviewed for applicability to the Lungmen ABWR Nuclear Island design. This review has been completed for the period of 1980 through late 1991. The conclusions from this review have been factored into the Lungmen ABWR Nuclear Island design.

During the performance of this review it was observed that some of the reviewed documents also had implications that are applicable to the balance of plant design, such as the ultimate heat sink design. These same documents will be reviewed again for consideration of their applicability to the balance of plant design. Any resulting design changes from this latter review will be documented in the FSAR.

In addition, as stated in PSAR Section 1.8.3 experience information from 1991 through June 1996 will be incorporated in the FSAR as appropriate. Further, any significant issues that arise after June 1996 will be reviewed and addressed on an individual basis, with the results of these reviews to be documented in the FSAR.

No PSAR revision is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-01-013

PSAR Sections: Original question referenced Ch. 1A, 1B, 1C, Follow-up question referenced Ch. 6.4

Question Date: November 22, 1997

PSAR Question:

1. When evaluating DBA for the habitable area of the control room, has the effects of radiation source or toxic gas source been considered ?

2. Is there independent system for the processing of non-condensable gas discharge of the system ?

3. When CRHA operates under the smoke-discharge mode, is the air intake still operating normally ? If it is, wouldn't it prolong combustion so its effect should be evaluated ?

PSAR Response:

1. The effects of radiation source and toxic gas have been considered and are provided in PSAR Sections 1C.2.40, 6.4, and 9.4.1. See also the response to Part 8 of question I-06-008 for proposed change to PSAR Section 6.4.

2. Noncondensable gases are removed and treated during plant power operation from the condenser by the steam jet air ejectors in conjunction with the gaseous waste system. Mechanical vacuum pumps that exhaust to the Turbine Building vent are used at the beginning of startup. Section 10.4.2 and 11.3.1 discuss these systems in greater detail.

   In the event of a DBA, the Standby Gas Treatment System as described in Section 6.5.1 treats and discharges air leakage containing halogens and particulates from the PCV.

3. During smoke removal mode, 100% outside air is supplied to the space and then is exhausted. No recirculation is allowed. The intent of this mode is to remove smoke from the CRHA to allow fire fighting measures to taken. The smoke must be exhausted to allow visibility in the CRHA so the fire can be fought. This exhaust air must be replaced by outside air.

   Typically hand held fire extinguishers and hose stations are used to manually fight fires. In electrical fires, the circuits that are causing the fire are de-energized to eliminate the source of the fire. Visibility is required for these fire fighting methods to be used. While the outside air does provide combustion air, this ventilation provides increased visibility so the fire can be fought effectively.

The following sentence will be added to the second to the last paragraph in PSAR Section 9.4.1.1.4: "Smoke is removed to maintain visibility to allow for fire fighting activities."

ROCAEC Review Comments:

SRP described specific prevention measures for toxic gases and explanation on emergency zone (see attachment) but in PSAR Section 6.4 these are lacking. It is suggested more explanations be provided

Further Clarifications:

GE will perform a hazardous chemical analysis for the control room to meet the requirements of SRP 6.4. The Control Room Emergency Zone consists of the South Corridor Panel Room (491), North Control Panel Room (492), Communication Equipment Room (494), Shift Technical Advisor's Room (495), Visitor Viewing Gallery (496), Switch & Tag Room (497), Shift Supervisor's Room (498), Main Control Room (499), Shift Supervisor's Conference Room (4910), Shift Supervisor Clerk Room (4911), Operator's Area (4912), Bathroom (4913), Storage (4914), Instrument Maintenance Room (4915), Process Computer room (591), East-Corridor (592), Perimeter Corridor (493), CRHA Supply "B" (691) and CRHA supply "C" (694). These will be listed in PSAR subsection 6.4 in the PSAR update submittal.

In order to do this analysis, the type, size and locations of potential leakage's need to determined. Next, a meteorology and probability analysis will be performed. Based on the analyses listed above the hazardous gas analysis would be then performed. Since the information required for this analysis is not available at the current stage of design, such an analysis will need to be submitted with the FSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-01-014

PSAR Sections: Ch 1A.2.30, 9.2.2, 9.2.11

Question Date: November 13, 1997

PSAR Question:

1. During LOCA, primary CTMT isolation valve of RBCW will close (see page 1A-27 section 1A.2.30(1), line 10). Please explain the effect on RMC ?

2. Is the last 2 lines of page 1A-27 describing that under Hot standby condition, RBCW must be operable to make sure the motor of RIP can survive ?

Response to Part 1:

After LOCA, a SCRAM occurs and all RIPs will runback to minimum speed within a few seconds after SCRAM has occurred.

Four or more RIPs will trip at Level 3 and will cause the independent RFCS logic to initiate an all RIP Runback to minimum speed.

At RPV Level-2 all RIPs will be tripped (4 at Level-3, 3 at Level-2, and the last 3 six seconds later after reaching Level-2.

The RBCW isolation valves close at RPV Level-1. By the time the water level reaches Level-1, the RIPs would have runback from rated speed to minimum speed and eventually tripped.

At LOCA condition and with RBCW isolated (RPV Level-1), the upper part of the motor will see higher temperature coming from the Vessel. This high temperature may damage the motor windings which can be replaced according to normal maintenance procedures.

RMC system is cooled by RBCW system. At RPV Level-3 and at Level-2, the RIPs may be tripped but RBCW system is still providing cooling water flow to the RIP heat exchangers. But at RPV Level-1, the RBCW for the drywell isolates and there is no cooling water to the RIP heat exchanger and that could lead to damage of the motor windings.

No PSAR modification is proposed in response to this question.

Response to Part 2:

In "Hot Standby" condition, it is expected that the Reactor is "shutdown" but at full temperature. Under this condition, the RBCW is available. It is possible to have RIPs operating at minimum speed to circulate water and prevent thermal stratification of the RPV bottom head.

No PSAR modification is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-01-015

PSAR Sections: Ch 1C.2.6 A-10

Question Date: November 21, 1997

PSAR Question:

1. According to Lungmen PSAR, single thermal sleeve design has been successfully applied since 1977. Then why the new design ? Is the structural integrity be impacted by the single sleeve in the future ?

2. Please explain :
(1) Is the double thermal sleeve a replaceable design ? what are the considerations ?
(2) How is it compared with the triple thermal sleeve used by GE domestic plants ?
(3) Would there be more restrictions on ISI execution ?
(4) Does ISI include consideration of IGSCC of crevice area when thermal sleeve is fixed, or there is no crevice ?

PSAR Response:

Response to Part 1:

Heat transfer through the thermal sleeve as a source of high cycle fatigue of the feedwater nozzle is discussed in the PSAR. Since the feedwater coming in through the thermal sleeve is at a lower temperature than the nozzle, the concern was that undercooled water could be shedding from the outside of the thermal sleeve and impinge on the RPV nozzle bore. Years of operating experience with welded single sleeve thermal sleeves used in BWRs in the US and abroad proves that this will not have any detrimental effect on the RPV nozzles. At the time of the ABWR design, this issue was still discussed and it was decided to apply a secondary thermal sleeve to preclude any high cycle thermal fatigue in the nozzle bore. Now, ten years after the double thermal sleeve design was introduced in the ABWR, the welded single thermal sleeves are still performing without problems and there is no reason to believe that the welded single thermal sleeve will produce any structural problems in the future.

If anything, the welded double thermal sleeve is considered to be an improvement of the welded single sleeve design.

Response to Part 2(1):

The double thermal sleeve used in the ABWR is not considered to be easily replaceable. If the thermal sleeve for some reason were to be replaced, the RPV nozzle safe end would have to be cut and the feedwater sparger removed together with the thermal sleeve. The same is the case for the welded single thermal sleeve. The excellent experience with the single thermal sleeve justifies the use of a welded design.

Response to Part 2(2):

Both the welded single thermal sleeve and the double thermal sleeve designs provide better protection against high cycle thermal fatigue than the triple thermal sleeve design as the welded designs do not have any leakage of cold water. Also, the triple thermal sleeve has a very complicated design that may require replacement after a number of years in service. The welded thermal sleeve designs are not expected to require replacement.

Response to Part 2(3):

ISI of nozzles with double thermal sleeves will not be more restrictive than that of nozzles with single thermal sleeves.

Response to Part 2(4):

The annuli between the inner and outer thermal sleeves and between the outer thermal sleeves and the nozzle bore have widths larger than 6.35 mm. Per the design spec which reflects GE's design practice, openings of a width larger than 6.35 mm are not considered crevices. Therefore, no consideration has been given to IGSCC in connection with ISI of the nozzles.

No PSAR modification is proposed in response to these questions.

ROCAEC Review Comments:

2. It is expected that the temperature will be improved with additional layer of thermal sleeve but it is still unclear how ISI plan will be carried out. Explanation should be given to how cracks at the intersection of the first and secondary thermal sleeve can be inspected.

Further Clarification:

2. Due to access restrictions, the welded connection between the first and second thermal sleeve can not be readily inspected, nor is it necessary to include this weld in the ISI plan. This welded connection is not classified as safety related, and; therefore, there has been no regulatory or code requirement to perform any ISI at this location. Even the recent documents produced by the BWRVIP inspection committee do not include any inspection requirements for the feedwater thermal sleeve and associated non safety related components for operating BWR plants. The function of the secondary thermal sleeve is to provide an additional thermal barrier to minimize that amount of thermal cycling on the inside surface of the feedwater nozzle. Postulated cracking at the thermal sleeve connection does not affect the ability of the secondary thermal sleeve to perform this function, and does not adversely affect any other

components which are safety related. Additionally, in the highly unlikely event that a postulated circumferential crack extends to a 360 degree through wall condition, the thermal sleeve would be fully captured and would not become a loose part. It would also continue to provide its intended thermal barrier function in this case.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-01-016

PSAR Sections: Ch 1A.2.30

Question Date: November 13, 1997

PSAR Question:

In page 1A-28, section 1A.2.30(3), line 8 to 10, the description was not sufficient to explain the condition when LOCA occurred during normal operation and RMISS was lost. Please elaborate.

Response:

Recirculation Motor Inflatable Shaft Seal (RMISS) subsystem is used only during plant maintenance outages. During normal plant operation, the inflatable seal is fully retracted away from the shaft and thus the shaft is not in contact with this seal. This subsystem is completely isolated when the pump is in operation and the pertaining valves are closed. Therefore, in case a LOCA occurred during normal plant operation, RMISS has no effect in its operation.

No PSAR modification is proposed in response to this question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-01-017

PSAR Sections: Ch 1C.2.15 A-42

Question Date: November 21, 1997

PSAR Question:

1. Nureg-0313 stated that the carbon content of IGSCC resistent material is below 0.035%, such as 304L, 316L, 304NG and 316NG. However, in Lungmen PSAR, 316 and 316L were mentioned and the carbon content of 316 is around 0.08% in general which far exceeds 0.035%. Please explain why it is used ?

2. If Nureg-0313 is to be followed, then the carbon content should be less than 0.035% but Type 316 s.s. has a carbon content far exceeds this value (0.08% in general). Please clarify.

3. In the Resolution paragraph, line 4, the 800k should be 700k.

PSAR Response:

Response to Part 1:

For piping GE will comply with the NUREG-0313 requirements for Category A material as it is defined in that document. The material will be further modified to have a carbon content even lower than required by the NUREG for Category A. In all cases for stainless steel piping and other stainless steel components exposed to high temperature reactor water (>93℃), material having a maximum of 0.02 % carbon will be used. This material will be certified as either Type 316 or 316L in accordance with ASME Code Section II (ASME does not include the NG designation nor does it have a separate designation for the modified 316L). The designation 316NG is a commercial description of the 316 type alloys having 0.02 % carbon that was created by GE in the late 1970s and may be interpreted to include 316L with respect to IGSCC resistance. The recognition by the U.S. NRC that 316L is accepted as a resistant material is documented in NUREG-0313. The NRC further clarified that 316NG was formerly called 316K which came from the K suffix designation in GE materials purchase specifications. The entire purpose of this NG designation was to commercially distinguish the material supplied by GE having 0.02 % maximum carbon from conventional 316, which is known to be susceptible to IGSCC in the welded condition. If the design calls for the strength properties of 316 material, as opposed to 316L, a minimum level of nitrogen is required to ensure the higher mechanical properties. However, in either case, the carbon maximum is 0.02 % and IGSCC resistance is the same. The only exceptions to this practice are for specialized components where fabrication or operating conditions demand an alternate material. Specific examples are control rod aborber tubes and in-core monitor components. Absorber tubes will be made of a special high purity, tantalum stabilized version of Type 304 called 304S for improved IASCC resistance. In-core monitors use a low carbon 304L material for improved weldability since no filler metal is used on such thin-walled parts. In all cases the intent of NUREG-0313, Rev. 2 will be met.

Response to Part 2:

See response to Part 1 above.

Response to Part 3:

Line 4 of the Resolution paragraph for Safety Issue A-42 on page 1C-34 of the Lungmen NPS correctly specifies a temperature range of between 700 and 1255°K, corresponding to 427 and 982°C.

No PSAR modification is proposed in response to these questions.

ROCAEC Review Comments:

1. Response to parts 1 and 2 : Nuclear Grade (NG) specification is indeed not in ASME and ASTM but the definition of low carbon content is widely known and referenced in many PSAR chapters. It is suggested that when measures were considered to deal with IGSCC problems and low carbon content (less than 0.02%) materials were used, they are denoted as NG materials to replace statements like "low carbon modified types 316 and 316L stainless steel" to avoid confusion.

2. Also, it was first mentioned in this section that "low carbon modified types 316 and 316L stainless steel" then it was stated that "stainless steel types 316 and 316L low carbon modified material". Does it mean that 316 and 316L, which are used to eliminate IGSCC in the Lungmen project, will have carbon contents less than 0.02% ?

3. The Question of part 3 was about errors in ch 1C.2.41 86 (page 1C-70) and not on ch 1C.2.15 A-42 which should be changed.

Further Clarifications:

RESPONSES TO ROC-AEC's PSAR QUESTIONS

1. We agree to the suggestion that the low carbon Type 316/316L materials (less than 0.02 %) should be denoted as NG materials; such as 316 NG and 316L NG.

2. Yes; Types 316/316L materials used to mitigate IGSCC in the Lungmen Project will have carbon content less than 0.02 %. This requirement applies to welded applications where IGSCC may be considered a concern.

3. We agree that the correct reference is 1C.2.41 86 (page 1 C-70). Also, we agree that in the resolution paragraph, line 4, the 800 K should be corrected to 700 K.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 01-018

PSAR Sections: Section 1.8

Question Date: December 24, 1997

PSAR Question:

Please explain why RG1.33, RG1.90, RG1.114, RG1.127, RG1.134 and RG1.149 in Table 1.8-20 are not applicable to Lungmen?

PSAR Response:

Regulatory Guide 1.90 "Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons" is not applicable to Lungmen NPS as its containment is a reinforced concrete containment, not a prestressed containment.

Regulatory Guide 1.134 "Medical Evaluation of Licensed Personnel for Nuclear Power Plants" is not applicalbe to Lungmen NPS because Taipower will follow the ROC-AEC's specific requirments for that purpose.

Following Regulatory Guides are applicalbe to Lungmen NPS and PSAR Table 1.8-20 will be revised accordingly:

(1) Regulatory Guide 1.33 "Quality Assurance Program Requirements (Operation)"

(2) Regulatory Guide 1.114 "Guidance On Being Operator At the Controls of a Nuclear Power Plant"

(3) Regulatory Guide 1.127 "Inspection of Water-Control Structures Associated with Nuclear Power Plants"

4) Regulatory Guide 1.149 "Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations"

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 01-019

PSAR Sections: Ch 1, Appendix 1D

Question Date: January 17, 1998

PSAR Question:

Regarding the design consideration of SBO, please clarify the following questions :

1. Please summarize and compare the different causes and its probability of causing SBO and total SBO probability for the First (Chinshan), Second (Kuosheng) and Lungmen NPS.

2. In the 1D.2.3.1.3 section on "SBO Events", it was mentioned that the swing EDG will be automatically or manually started when LOOP signal is received. Is it designed as automatic or operator action is required to connect it to the Bus ? Please clarify.

PSAR Response:

1. The Chinshan, Kuosheng, and Lungmen nuclear power plants apply a defense-in-depth philosophy to prevent SBO by utilizing multiple, diverse and redundant electric power sources and distribution circuits. Key differences identified between the three plants that may impact the SBO probability include:

   1) Level of redundancy of the emergency diesel generators (EDGs) and Alternate AC Power Source (AACS)
   2) Capability of the nuclear power unit to accept a load rejection without scram, and thus provide power from the main generator for house loads in case of loss of offsite power
   3) Accessibility of electric power from one unit to another unit on site
   4) Type and redundancy of the Alternate AC Power Source (AACS) and the time needed to start loading the AACS.

   The redundancy and diversity of the electric power sources and distribution circuits in the three plants mean that an SBO event is possible only if specific combinations of multiple failures occur. These combinations include the following individual failures:

   1) Loss of offsite (normal and reserve preferred power supply) transmission network.
   2) Loss of the plant switch yard,
   3) Failure of the high voltage circuits from the switch yard to the unit and reserve transformers.
   4) Unavailability of electric power from the main generator of the unit or another unit on site,

5) Unavailability of the EDGs,
6) Unavailability of the AACS of the unit or another unit on site, and
7) Inability to recover an electric power source before safety limits are reached.

The underlying causes of the above SBO attributes include those due to internal equipment failures, failures of interfacing or supporting equipment, unavailability due to outage for test and maintenance (T&M), or human error. They also include independent as well as common cause failures (CCF) of redundant channels and electric power sources. A detailed fault tree model of electric power unavailability that accounts of such underlying causes for the Lungmen NPS can be found in Attachment AA of Appendix A, Probabilistic Risk Analysis. For the purpose of this comparison, simplifying assumptions have been made to estimate the probability of top-level causes of the SBO and total SBO probability, with main focus on key differences between the three plants.

The analysis uses simplified probability models for estimating the SBO probability and its main contributors. Two definitions of the SBO event have been used in the analysis:

1) The SBO definition in 10CFR50.2 and Regulatory Guide 1.155, where no credit is taken for the AACS, and
2) SBO definition generally used in PRA where credit is taken for the AACS.

The probability estimates are based on the following simplifying assumptions:

- No credit is taken for recovery of any electric power source
- No account has been made for failure contribution by support systems of the Emergency DGs (EDG) or Alternate AC Power Sources (AACS)
- No account has been made for failure of the high voltage circuits from the switch yard to the unit and reserve transformers
- A mission time of 8 hours is assumed for the EDG and ACCS operation.

Other assumptions used in the analysis are indicated in the attached Table 01-019-1.

Table 01-019-1 contains a comparison of the key features that impact the SBO probability, the SBO contributors, and the probabilities obtained using the above assumptions. For each of the SBO definitions given above, two Lungmen SBO probability estimates are presented; with and without electric power transmission from the main generator of one unit to another given a SBO event in the latter unit.

The results in the table point to the following conclusions:

- The Lungmen SBO probability is lower than those of the other plants due to the Lungmen higher level of EDG redundancy (3 EDGs/unit for Lungmen vs. 2 EDGs/unit for the other plants) and Lungmen capability to accept a load rejection event without scram. The Lungmen SBO probability is further reduced (by about a factor of 2) if the power from the main generator of one unit is supplied to the other unit in case of SBO in the latter unit.

- Due to the high EDG redundancy in Lungmen, independent failures contribute a small fraction (~10%) of the EDG failure probability. In contrast, EDG failure probability contribution for Chinshan and Kuosheng is more-or-less equally divided between independent failures, CCF, and failures involving a T&M unavailability of one EDG.

2. The swing EDG connection to a division is manual, as discussed in PSAR Chapter 8, Sections 8.1.4 and 8.3.1.1.7(9). We agree the last sentence of 1D.2.3.1.3 should be clarified. Therefore, it will be replaced with the following two sentences: "A LOOP signal automatically starts the swing EDG whether or not it is configured to a Class 1E bus. The configuration itself is manual, and can be performed at any time (i.e., before or after LOOP) by plant operators to a selected Class 1E bus using appropriate procedures."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 01-020

PSAR Sections: Ch 1

Question Date: December 26, 1997

PSAR Question:

1. In the Appendix 1D.2.2.2 Section, under SBO Design Basis, it was mentioned that "The Lungmen NPS design will confine the SBO duration to (to be provided in FSAR) minutes or less with the use of the AAC power source (to be confirmed)".

2. In the 1D.2.3.1.3 section on SBO Events, it said "Swing EDG will be available for bus connection within (to be provided in FSAR) minutes".

3. Please explain why those information in the above parenthesis have to wait until FSAR?

PSAR Response:

There are several reasons why this information will not be available until the FSAR. First, as indicated in the response to ROC-AEC Question 01-019, the configuration of the swing diesel generator (SDG) is a manually controlled action. Therefore, the time for operators to perform this action is dependent on Emergency Operating Procedures (EOPs) and the site security plan, which have not yet been established for Lungmen NPS.

Second, the process through which the operator configures the SDG is dependent upon the logic design for the SDG bus, which was not available for the PSAR, but will be available with the FSAR. For example, the interlocking arrangement to assure that only one bus can be connected to the SDG at any one time, while still maintaining licensing separation requirements, is still in the design phase. It is not yet established whether the operator will be able to control the configuration directly from the main control room (lesser time, but higher risk of multiple feed), or whether he will need to physically go to the Auxiliary Fuel Building (AFB), where the SDG is located, to operate these interlocks (longer time but lower risk of multiple feed).

In summary, the overall time necessary to perform these activities must be determined based on documents and procedures which are not available until the final design.

No change to the PSAR is planned in response to the above question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 01-021

PSAR Sections: Ch. 1

Question Date: February 20, 1998

PSAR Question:

Please provide the differences in design and its impact on safety, operation and maintenance among Lungmen, K-6/7 and USABWR.

PSAR Response:

Because of KK-6/7 owner's reluctance to release its design information and the comparison to other operating ABWRs are not considered to be pertinent as the other ABWR plants were constructed and licensed according to regulations in another country, the response to this question will focus on a comparison of the Lungmen NPS and USABWR design. The Lungmen design follows very closely the USABWR Certified design. Lungmen design changes have only been made where necessary to comply with differences in ROC regulations and TPC's Bid Specification. The residual top level design differences are summarized in the attached table. A discussion of their impact on safety, operation and maintenance follows.

1. Lungmen's U.S. and ROC codes & standards vs. Certified Design's U.S. codes & standards

   The primary difference between the two designs is the use of the ROC Fire Protection Code for the Rad Waste Building, Hot Machine Shop and Turbine Building for Lungmen. Of course on the administrative side, two step licensing will be applied for Lungmen while U.S. regulations also allow for one step licensing.

2. Lungmen's 40 year plant life vs. Certified Design's 60 year plant life

   There will be no effect due to this difference during the 40 year licensed life.

3. Lungmen's 0.4 SSE and 0.2 OBE vs. Certified Design's 0.3 SSE

   Due to Lungmen's site specific requirements the design has been enhanced to maintain the original low seismic risk goals. An OBE design requirement has been retained consistent with the two step licensing process.

4. Lungmen's ocean cooled site vs. Certified Design's cooling towers and spray pond

   The availability of the ocean for Lungmen's cooling needs increases the plant's power generation reliability and Ultimate Heat Sink reliability.

5. Lungmen's 110% bypass capacity vs. Certified Design's 35% bypass

Lungmen's 110% bypass capacity decreases its Station Blackout risk and allows the plant to accept a full load rejection without reactor shutdown. The 110% bypass system will also not scram on a turbine trip, with the exception of high condenser vacuum indicating decreasing heat removal capability (loss of heat sink). No SRVs will open on either a full load rejection or turbine trip. With the Certified Design's 33% bypass system, reactor will scram and SRVs will open upon a load rejection or turbine trip above about 33% power.

6. Lungmen's power cycle vs. Certified Design's power cycle

   The power cycle design differences were selected for Lungmen to make the Lungmen design and equipment more consistent with existing TPC's operation and maintenance practices and procedures. This consistency across TPC's facilities lowers the risk of human error.

7. Lungmen's electrical power distribution system vs. Certified Design's electrical power distribution system

   The electrical power distribution differences were selected for Lungmen to provide sufficient power for Lungmen's specific electrical loads and to make the Lungmen design and equipment more consistent with existing TPC operation and maintenance practices and procedures. This consistency across TPC's facilities lowers the risk of human error.

   In summary, each design achieves the applicable safety standards and addresses individual operation and maintenance practices.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 01-022

PSAR Sections: Ch. 1D

Question Date: February 3, 1998

PSAR Question:

1. In Table 1D-1 (page 1D-18) statement, coping analysis would have to be performed if 10CFR50.63 requirement is followed. Please describe the contents of such an analysis.

2. As it is described in the certified design material (page 2.4.4-10) for a standard ABWR plant, the RCIC system can operate without AC power for a period of at least 2 hours. Please identify the design improvements made to ABWR and Lungmen NPS so that the statements can be justified as given on page 1D-8 "Lungmen NPS can withstand an SBO with failure of the SDG without core damage or loss of containment integrity for a period of 8 hours."

3. In page 1D-12 the following statements are given "The suppression pool temperature exceeds its design value of 97.2°C after 8 hours." Please explain during SBO, what is the design basis for the suppression pool design ? What are the effects when the suppression pool temperature exceeds its design value (97.2°C) ?

PSAR Response:

1. A two hour AC independent coping analysis for Station Blackout will be performed for Lungmen NPS, with the results to be presented in the FSAR. This analysis will assume the occurrence of Station Blackout in one unit, with the remaining unit, at a minimum, retaining the necessary safety-related AC power to achieve safe shutdown. For the one unit assumed to experience station blackout no credit will be taken for Alternate AC power for the two hours of assumed blackout in the analysis. This means that all three EDGs of that unit and the SDG will be assumed to be unavailable for the two hour period.

   This two hour coping analysis will take credit for RCIC operation to maintain RPV level. RPV pressure will be controlled via SRVs. The analyses will demonstrate that the batteries are sufficiently sized to provide DC power for this two hour period, and secondary containment analyses will demonstrate that temperatures remain appropriate for RCIC operation. RPV water level, RPV pressure, suppression pool temperature and containment pressure analyses will all show that these parameters remain within appropriate limits.

   Since the SDG will be started and aligned to the appropriate electrical bus within (to be determined later) minutes after the start of the Station Blackout event, the above analysis will be more than sufficient to satisfy the requirements of 10CFR50.63 for a coping analysis.

2. Both the certified ABWR design and Lungmen have been improved relative to earlier vintage BWRs regarding the operating capability of RCIC under SBO conditions. These design improvements are summarized in Section 1D.3.1. However, there is no difference between the certified ABWR and Lungmen RCIC requirements. As stated in PSAR Chapter 5, page 5.4-18, the Lungmen RCIC has a 2 hour design basis SBO capability, and an 8 hour non-design basis capability. The differences between the 2-hour and 8-hour capabilities are summarized below.

   The 2 hour capability will be demonstrated using conservative design basis assumptions and analysis, and environmental qualification of the as-built RCIC and auxiliaries. A two-hour SBO will be the design basis for establishing the capability of the 1E DC power, RCIC water supply sources, and RCIC equipment room temperature. The analysis will use conservative decay heat for assessing the availability of RCIC water sources for 2 hours. Conservative heat sources and heat transfer calculations will be used to estimate the RCIC equipment room temperature. The environmental qualification of the as-built RCIC will be based on design-basis LOCA environmental conditions to establish RCIC reliability for the 2-hour design basis duration.

   The 8 hour capability will be demonstrated using realistic, best estimate assumptions and analysis methods. The basis for this capability is an SBO for 8 hours. Capability of the 1E DC power sources will be assessed on the basis of 8-hour of RCIC operation. Realistic decay heat load will be used to assess the water sources capability. The analysis to estimate the RCIC equipment room temperature will use realistic heat sources and best estimate heat transfer analysis that may take credit for potential mechanisms of heat removal from the room. These analyses will establish RCIC capability to provide sufficient reactor cooling water for the assumed 8-hour SBO.

3. The suppression pool design temperature value of 97.2°C is based on the NPSH requirement of the ECCS pumps during design basis LOCA events, and this NPSH requirement does not take credit for pressurization of the wetwell air space during LOCA conditions. Structural integrity of the suppression pool design is determined by combined dynamic (LOCA and SRV) and thermal loads (due to pool temperature). In comparison, the dynamic loads are found to be substantially more severe than the thermal loads, from the containment structure design standpoint.

   During SBO, there are no LOCA dynamic loads, and the SRV dynamic loads are expected to be substantially lower than those considered in the load combination for the design basis accident event. The design basis event load combination considers dynamic loading due to all 18 SRVs, compared to only dynamic loading due to few SRVs opening during SBO. Therefore, during SBO, the suppression pool is expected to experience a substantially lower loading condition than that considered for the design basis accident event.

There exists no SBO specific design basis for the suppression pool temperature. However, based on PRA analyses, suppression pool temperature reaches to a maximum value of 430 K (157°C), and the suppression pool is limited to a maximum pressure of 0.72 MPa by the COPS (Containment Overpressure Protection System), as reported on page AJ.1-19 of Attachment AJ to Appendix A of the Lungmen PSAR.

Therefore, in view of the above, it can be concluded that the suppression pool temperature value of 112.2°C after eight hours during SBO is expected to have no adverse impact on containment structure integrity.

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

ROCAEC Review Comments:

Item 2, please show what improvements have been made to RCIC compared with conventional RCIC.

Further Clarification:

The design of the RCIC for Lungmen has been improved with the use of a mechanical feedback for flow control. It does not use the electronic feedback to adjust the turbine steam control valve. This uses less electrical power from the station battery. It is also a more direct and immediate feedback which improves the reliability of the system to achieve cooling water injection. The pump and turbine are capable of operation without any electrical power. The pump and turbine are in a single case eliminating the need for integration of two or more components. The pump and turbine are lubricated by the water that is pumped. This eliminates the need for a separate oil system and its associated cooling and filtering equipment. The new design is relatively simple and very reliable.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 01-023

PSAR Sections: 1C.2.2

Question Date: March 27, 1998

PSAR Question:

1. When water level in a heater is too high and emergency dumping is necessary, water hammer would possibly occur as a result of hot water flashing at the outlet of the emergency discharge piping. Experiments done at MIT confirmed this possibility (Note 1). Please explain how it can be avoided.

   Note 1. Sweeney, E.J., "Water Hammer Transients in Two-Phase Flow" M.S. Thesis, Department of Mechanical engineering, MIT, 1998.

2. PSAR Chapter 16 Section 16B.3.5-3 required that keep fill system be provided for ECCS discharge line. But in Chapter 6, page 6.2-141, only RHR was mentioned to have keep fill pump. Other than that, Table 8.3-1 only listed RHR fill pump for EDG load. Please clarify whether HPCF and RCIC also have keep fill system ? Furthermore, which systems have installed keep fill system ? and whether the pumps of this keep fill system get their power from the emergency power source ?

PSAR Response:

1. Based on the S&W design, the feedwater heater emergency drain control valve is an air operated globe valve with fail-open valve operator setup. The opening of the valve is controlled by the pre-set water level in the feedwater heater shell. A typical 16-inch control valve of this type will take about 30 seconds to move from fully open to fully closed position. Thirty seconds is far beyond the time needed to be classified as "rapid closure" operation for water hammer consideration. In the MIT June 1989 M. S. Thesis of Master of Science " Water Hammer Transient in Two -Phase Flow", by Edmund J. Sweeney, 0.001 second was used as the valve closure time in its "RELAP5" computer runs to simulate the experiment.

   S&W has not had any water hammers problem in our feedwater heater emergency drain piping design in the past and have no reason to expect any water hammer problems in Lungmen's feedwater heater emergency drain piping.

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

2. Prior to initiation, the HPCF and RCIC are maintained in full condition by the CSTF pumps (non-Class 1E powered) via the keep fill lines and also by the CST elevation head through the pump suction lines.

After emergency initiation with full lines, the HPCF runs continuously in injection or min-flow modes until it is no longer needed as determined by plant operator. It does not require a separate keep fill system following its initiation.

The RCIC has an installed keep fill pump to maintain keep fill function after RCIC initiation. After emergency initiation the RCIC will cycle on and off in response to RPV level signals. The keep-fill pump is provided for the RCIC to maintain RCIC lines full during the period after RCIC emergency initiation, when the turbine-pump is turned off by high reactor water level. The RCIC keep fill pump is powered by an Class 1E DC power supply.

In summary, the RHR and RCIC have dedicated keep fill pumps which are powered by Class 1E sources.

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

ROC-AEC Review Comments:

Response to item 2 not accepted for the moment.

1. HPCF will automatically trip when Level 8 is reached. If it is needed when water level drops again, it should be clarified whether water hammer phenomenon will occur (since CSTF pump is supplied from non-class 1E power source which could fail during LOCA).

2. Since RCIC has Keep fill pump and supplied from class 1E power source, it is suggested that Table 8.3-1 be modified to reflect these facts.

Further Clarification:

1. When level 8 is reached the injection valve is closed and the HPCF pump will continue to run in the minimum bypass mode. This will maintain the water level to preclude water hammer.

2. RCIC, including its keep fill pump, is powered by the Class 1E 125 V DC System. Therefore, the RCIC is not a EDG load and is not included in Table 8.3-1.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 01-024

PSAR Sections: 1A.2.9

Question Date: April 20, 1998

PSAR Question:

In the last sentence of this subsection, the following statement is given "The Lungmen NPS does not require SRV testing with liquid under low pressure conditions associated with this event as required in past BWRs."

Please explain this by comparing the approaches taken by Kuosheng and Lungmen NPS respectively.

Response:

One of the improvements made to the ABWR was to equip each of the three RHR loops with its own shutdown cooling suction piping connection directly to the Reactor Pressure Vessel. By comparison, earlier vintage BWRs such as Kuosheng only had one RHR suction line connection to the RPV. The single failure protection backup to the valve on this suction line for these earlier BWRs was the Alternate Shutdown Cooling Mode. For plants like Kuosheng, the Alternate Shutdown Cooling Mode is accomplished by flooding the RPV at low pressure, opening the SRVs and running the RHR in the injection mode while passing the flow through the RHR heat exchangers. For Lungmen, the single failure protection backup to the failure of one RHR suction line valve failing to open is to open one of the other two RHR suction line valves.

Thus, for Lungmen there is no need to operate the reactor in Alternate Shutdown Cooling. Accordingly, there is no mode of operation for Lungmen that will discharge water through the SRVs, and no testing of such conditions is required.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 01-025

PSAR Sections: Chapter 1, Appendix 1A

Question Date: May 12, 1998

PSAR Question:

The following statements are provided in page 1A-33 of the PSAR. "The MSL tunnel area is monitored for high radiation levels and for high ambient temperatures that are indicative of steam leakage. The Turbine Building is also monitored for high area ambient temperatures for MSL leakage. The resulting action causes isolation of the MSIVs and subsequent shutdown of the reactor." Please clarify whether the high radiation level will cause the isolation of the MSIVs or not.

Response:

Isolation of the MSIVs in response to steam leakage in the MSL tunnel area is provided by a high ambient temperature signal. Thus, the MSIV isolation signal for the MSL tunnel area is similar to that for the Turbine Building.

As explained in the response to question 07-002, the MSIV isolation signal from the Main Steam Line Radiation Monitor has been eliminated for Lungmen as well as for many operating GE BWR's. The basis for the elimination of the MSL high radiation isolation signal is documented in NEDO-31400A "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor". The remaining high ambient temperature signal provides a fully redundant means of isolating the MSIVs in the event of a MSL steam leak.

Accordingly, page 1A-33 will be revised to read. "The MSL tunnel area is monitored for high ambient temperatures that are indicative of steam leakage."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： 01-026

問題章節(PSAR Section)： 1C

初提日期(Question Date)： 1998.5.13

問題內容(PSAR Question)：

The following statements are given in PSAR page 1C-93. "The acceptance criteria for the resolution of Issue 145 is to demonstrate compliance with the maintenance rule, 10CFR50.65." "A program that complies with 10CFR50.65 will be developed and implemented." The maintenance rule 10CFR50.65 is currently not a licensing requirement of ROC-AEC. Will TPC implement the program on a voluntary basis?

問題答覆(Responses)：

Yes.
For the details please refer to responses to question N-01-009.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： I-02-001

問題章節(PSAR Section)： 2.5.1

初提日期(Question Date)： 1997.11.17

問題內容(PSAR Question)：

第2.5.1.2.2節：請補述屈尺斷層以南枋腳層及媽岡層之特性。

問題答覆(Responses)：

補充說明如下：

(1)以下文字將補插入龍門安全分析報告第2.5.1.2.2.2節(page 2.5-18) 第二段文字之後，如附件所示：

"As shown in table 2.5.3, Makang Formation is correlatable to Wuchishan Formation. According to Huang and Liu (Reference 2.5-54), Makang Formation is distributed in the southern part of Chuchih fault. It is an alternation of fine- to medium-grained pale grey sandstone and dark shale. The sandstone is rather compact and competent. The thickness of this sandstone is about 2 meters in general, however, some bedding with thickness more than 5 meters in some local places usually is forming a hog-back topographic expression. The shale part was subject to slight metamorphism and appeared as an indurated argillite. The total thickness of this formation is about 370 meters."

(2) 以下文字將補插入龍門安全分析報告第2.5.1.2.2.3節(page 2.5-19) 第一段文字之後，如附件所示：

"Fangchiao Formation is correlatable to the Mushan Formation and is distributed in the southern part of Chuchih fault. It is mainly the white, fine-to coarse-grained sandstone and grey shale in alternation, occasionally with some thin interbeds of siltstone and laminated coal seams. The sandstone is thick-bedded and cross-bedded in appearance, representing the deposition in shallow shelf to parallic environments. The exposed thickness of this formation is about 300 meters.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： I-02-002

問題章節(PSAR Section)： 2.5.3

初提日期(Question Date)： 1997.11.17

問題內容(PSAR Question)：

一、表2.5-7：地震資料蒐集時間僅至1989年，請予更新補充。

二、第2.5.3.4節、第2.5.3.5節及第2.5.3.6節：所稱參考文獻2.5-21應為2.5-5之誤。請改正。並請摘述相關重要數據如定年資料、遙測分析、微震分析等之具體結果及圖表之說明。

三、第2.5.3.3節及第2.5.3.5節：對於地震資料均指引至第2.5.2節，然事實上該節並無具體摘述相關資料，請補充說明。

四、第2.5.3.7節：內容請參考10 CFR 100 App.A III (j)(k)要求請詳細說明。

五、第2.5.3.8節：請補充討論圖2.5-8中，燦光寮斷層之評估結果。

六、第2.5-23頁第15行"Past two years"寫法不妥，請明確說明年代。

七、表2.5-4，地下水資料僅至1982年，請更新資料。

問題答覆(Responses)：

一、自1989年2月迄1998年2月止之10年期間，於核四廠址半徑10公里範圍內，共蒐集到71次地震規模1.0以上之有感地震紀錄，其中最大之兩次地震分別為M4.44及M4.47（詳附件一之附圖一、附表一）；此遠低於應用於計算核四廠址設計基準地震值所使用地震規模7.3之最大歷史地震紀錄。

依據附圖一之資料，並無顯示由密集震央所構成地質線形構造之現象，此說明該地區並沒有活動斷層存在。此亦與PSAR內容（附件一之附圖二）所述結論相同。

綜上所述，經蒐集評估1989年迄今之地震資料，並未影響PSAR所述設計值，亦證實廠址安全性無虞。

二、第2.5.3.4節、第2.5.3.5節及第2.5.3.6節所稱參考文獻2.5-21確係2.5-5之誤，將照指正改正。至於相關重要數據如定年資料、遙測分析、微震分析等之具體結果及圖表之說明，均包含於前已獲原能會審查通過之報告："Fourth Nuclear Power Plant, Reevaluation of PSAR Section

2.5.2"(Reference 2.5-59)，請參考。

三、有關第2.5.3.3節及第2.5.3.5節對於地震資料，因資料係沿用前已獲原能會審查通過之報告："Fourth Nuclear Power Plant, Reevaluation of PSAR Section 2.5.2"，故此處未再予以摘述，但龍門PSAR第2.5.2節中已說明並將該報告列為參考文獻(Reference 2.5-59)。

四、PSAR第2.5.3.7節將改寫如下，畫底線之文字係新增加之部份：

"As mentioned in Section 2.5.3, northeastern Taiwan is characterized by the development of an imbricate thrust belt. The early phase of faults study consisted of 5 months of field work to gather as much surface and subsurface data as possible. Following this effort, additional field work at the phase I site study <u>in 1981</u> was satisfactorily completed. <u>The fault studies performed in the field work including trenching, aditting, inclined hole drilling, and satellite and air photo remote sensing are to identify fault existence and the age relationship. Six important faults namely Aoti fault, Shuanghsi fault, Kungliao fault, Fanchiao fault, and Wentzukeng fault were thoroughly studied. All information collected by the fault studies indicated that the Fanchiao fault is the youngest fault and the age of it is older than 30,000 years and possibly older than 100,000 years by C-14 age dating. Therefore, there are no capable fault in the vicinity of the site. In 1994, some trenches excavated in the vicinity of the proposed reactor site area for the study of re-examination and re-appraisal of the site conducted by Geological Society of China confirmed the former conclusions (Reference 2.5-7). In addition, there are no linear features indicated by the epicenter to suggest the active faulting near the site. Therefore, the detailed investigation of the regional and local geologic and seismic characteristics of the site demonstrated no need to design for surface faulting.</u> The <u>detailed</u> evaluations and overall conclusions are discussed in the following section."

五、

(1) PSAR第2.5.3.8.6節將改寫如下，畫底線之文字係新增加之部份：

"There are numerous other small faults throughout the area. Many of these are very old and are basically intraformational, having rocks of the same age on either side. <u>One of these minor faults, such as Tsankuangliao fault is an inferred fault extending in north-west direction in the northwest of Aoti village. According to Huang and Liu(1968), it was recognized by severe variation of the attitude of strata. It was almost covered by dense vegetation and volcanic detritus. No exposure of this fault was found. Because it was truncated by Aoti fault and probably a right lateral strike slip fault, and was evaluated not a capable</u>

fault according to USNRC guideline. Where these faults like Tsankuangliao fault intersect the more important faults, they are invariably terminated. Many of these smaller faults are inferred, and their locations are uncertain. None of them is significant to the siting of the plant."

(2) PSAR第2.5.3.8.7節在page 2.5-32之第二段文字將改寫如下，畫底線之文字係新增加之部份：

"Some trenches, No. 9 and No. 11, demonstrated that the terrace and colluvial deposits overlying the Fangchiao fault are undisturbed by faulting and are therefore younger than the age of last movement on this fault. The last movement on the Fangchiao fault, the youngest fault in the area, is therefore older than 30, 000 years and possibly older than 100,000 years. In 1994, Professor Y. Wang, the former president of Geological Society of China, and some other scholars concluded on their studies of the site that the up-to-date theory and experiment prove that the bedding-plane fault observed in the trenches and bore holes are actually, prelude of , or concurrent with the relevant large scale thrust under compressive stress field. Since the large scale thrust in the site or in the adjacent area were evaluated as non-active, those disturbances or fractures zone encountered in exploration program are logically interpreted to be no-active. (Reference 2.5-7)"

六、第2.5-23頁第15行"Past two years"將修改為:

"the year of 1980 to 1982 in phase I and phase II exploration"

七、核四廠址地下水水位設計值為El. 12m（地表），由於廠區內並無泉湧，故此水位已涵蓋可能之觀測值。因此維持PSAR Table 2.5-4之水位觀測記錄不做更動，不會實質影響PSAR有關地下水章節之陳述與結論的保守性。檢附民國八十三年七月到八十六年九月之廠區地下水觀測記錄（詳如附件二），請參閱。

八、另本公司於答覆本問題時，發現第二章部份章節有打字錯誤或已漏資料處，擬一併修改如附件三，詳述如下：

(1) page 2.5-20,上面算來第二行末增加一個參考文獻Reference 2.5-7

(2) page 2.5-27, section 2.5.3.3第二段第一行，8km修改為5miles

(3) page 2.5-27, section 2.5.3.5第二行，8km修改為5miles

(4) page 2.5-55, Table 2.5-1,第一列第三欄，5km修改為5miles

(5) page 2.5-57, Table 2.5-3第一列第二欄，Fangchia m修改為Fangchiao Fm，Makang m修改為Makang Fm

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： I-02-003

問題章節(PSAR Section)： 2.4.2

初提日期(Question Date)： 1997.11.05

問題內容(PSAR Question)：

一、文中提到由於歷年之資料不夠（只有間斷性且在民國七十二年以前之資料），無法直接進行頻率分析，雖然核四場址水文是以石碇溪之資料為基準，但從雙溪水文站觀測資料，可知最大洪水量為民國七十八年之1,020cms（見附表），較文中提到雙溪之625cms為大，所以鄰近之石碇溪可能有更大洪水量產生，故表2.4-1分析之各頻率洪水量，已與觀測資料不符，是否會對廠址各項機組之高程設計有影響，請澄清

二、請補充民國七十二年以後之資料，以更新本節相關迴歸週期之頻率洪水資料，並評估是否仍可以12.0m MSL做為設計洪水位。

問題答覆(Responses)：

一、雙溪與石碇溪分處龍門廠址之南、北方，由於山岳地形區隔，該兩條河川屬不同水系。依歷年(1981~1996)在雙溪與石碇溪觀測到之流量(詳見附表一、二）顯示彼此間之洪水量無直接關連性，石碇溪亦未如雙溪出現更大之洪水量。而主要排洪渠道係依可能最大的降雨量(PMP)設計，PMP之設計值為624~930mm/hr，查龍門廠址歷年(1980~1997)觀測到之最大降雨量為1989年之79mm/hr（詳見附表三），與分析所用之資料並無明顯之偏離，相較於設計所用降雨量之保守餘裕度，仍無排洪功能上之疑慮。

二、廠址高程設計之主要考量因素為海嘯、颱風對海水位之影響，龍門廠址設計高程採EL. 12M係取決於颱風時之海水上升高度。故前述問題不影響廠址高程之設計。

編號(Track Number)： N-02-004

問題章節(PSAR Section)： 2.1.3.1

初提日期(Question Date)： 1997.11.19

問題內容(PSAR Question)：

本節係敘述距離反應器半徑50公里之人口分佈情形，表2.1-3提出1991年於不同距離範圍內之人口統計及於2034年人口密度之預估。
但表中於0-2^{km}至0-30^{km}預估2034年之人口密度反而較1991年之普查數據較少，其中0-2^{km}至0-10^{km}人口密度幾乎減半，似乎與人口逐年成長之原則相違，請詳細解釋之，並近年內有無再作人口普查，調查所得之數據為何？

問題答覆(Responses)：

對於未來人口成長之推估，本報告以民國三十九年至民國七十九年間人口成長趨勢作為推估之根據。由於人口成長率受自然環境、經濟條件、交通便利、醫藥衛生及教育程度等因素所影響。台灣地區由於教育普及、推行家庭計畫等措施、最近十多年來、整體出生率呈現顯著降低趨勢。至於廠址半徑五十公里內之人口變化則由於各市鎮的社會經濟條件不同，城市化程度不一，以致各市、鄉、鎮間的成長趨勢互異，而且在不同時間亦有不同之成長形態。依人口成長率之不同約可歸納為四類：

一、人口成長快速地區

此區人口成長快速，多屬社會性人口流入地區，包括台北市、板橋市三重市、永和市、中和市、新莊市、新店市、汐止鎮、土城市、泰山鄉、五股鄉、蘆洲鄉等地，主要集中在台北都會區。

二、人口成長緩和地區

此區人口成長穩定，包括基隆市、三峽鎮、淡水鎮、深坑鄉及羅東鎮等地區。

三、人口成長緩慢地區

此區人口外流現象嚴重，但是人口數仍呈增加，包括瑞芳鎮、坪林鄉三芝鄉、石門鄉、八里鄉、貢寮鄉、金山鄉、萬里鄉、烏來鄉、宜蘭

市、蘇澳鎮、頭城鎮、礁溪鄉、五結鄉、壯圍鄉、員山鄉、冬山鄉及三星鄉等地，主要分布於蘭陽平原一帶，其中瑞芳鎮、坪林鄉、石門鄉、貢寮鄉、蘇澳鎮及三星鄉近年來人口呈現減少現象，有趨向於人口成長衰退地區之趨勢。

四、人口成長衰退地區

此區人口成長率為負值，人口外流現象最為嚴重，包括石碇鄉、平溪鄉及雙溪鄉等地，大多位於台北縣貧瘠、偏遠之山地地區，交通較為不便，經濟發展落後。

在距離廠址0-10 Km範圍，主要涵蓋區域包括貢寮鄉、雙溪鄉等人口成長衰退地區及少部分頭城鎮、瑞芳鎮等人口成長緩慢地區。總和估算0-10 Km範圍內人口外流嚴重屬人口減少地區。因此人口密度幾乎減半。0-30 Km範圍內包括貢寮、雙溪、平溪、石碇、坪林、瑞芳等人口減少鄉鎮及頭城鎮屬人口成長緩慢地區，基隆市屬人口成長緩和地區，只有汐止鎮位於人口成長快速地區。由PSAR表2.1-1及2.1-2比較可知在0-20Km以內人口均有減少現象，僅20-30Km環狀地區內人口增加。因此，總和估算0-30Km範圍人口密度略為減少。

近年本公司並未再正式大規模進行龍門核電廠附近人口調查，但計畫於FSAR之前考慮重做人口調查以更新現有數據。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-02-005

PSAR Sections: Ch 2.2.1.2.2

Question Date: November 20, 1997

PSAR Question:

Tab water supply is a necessary resource for nuclear power plants' operations but no mention was made in the PSAR as to if the supply will be adequate during the dry season? Is there any storage facility or dam at the water source? Please explain.

PSAR Response:

Please see PSAR Subsection 2.11 "Low Water Consideration" for a discussion on how LNPS 1 & 2 has been designed to handle periods of low water levels. Specifically Subsection 2.11.5 states the following:

"The fresh water supply requirements for the two units is about 0.05cm which is less than the minimum recorded flow at the Shuang Chi hydrometric station (Table 2.4-12). However, to cope with the low water flow resulting from the drought, the raw water reservoir of 120,000m3 capacity will be constructed at the Yenliao site (Figure 2.4-4) to ensure all plant raw water supply requirements are maintained."

No changes in PSAR Section 2.2.1.2.2 are required.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-02-006

PSAR Sections: Ch 2.5.4.8

Question Date: November 19, 1997

PSAR Question:

The description for soil liquefaction potential in this section is too brief. Please supplement with the following information:

1. Are all safety-related structure foundations founded on rock? Please provide profiles for Units 1 and 2, indicating ground water table, structure foundations and rock structure, and NS and EW section views.
2. Is there a soil liquefaction potential for shallow structures? If soil settles due to liquefaction, what would be the consequences to the safety-related structures?

PSAR Response:

1. All safety-related foundations are founded on rock. Section views of the NS and EW directions showing safety-related foundations, rock structure, and water table will be prepared and inserted into the PSAR. These new sections views will be labeled "Figure 2.5-12a Geologic Section A-A (NS Section)","Figure 2.5-12b Geologic Section B-B (EW Section)", and "Figure 2.5-12c Geologic Section C-C" and will replace the current Figure 2.5-12.

   The following statement will be added to the end of PSAR Subsection 2.5.4.8: "The section views of the NS and EW directions showing safety-related foundations and rock structure are presented in Figure 2.5-12. The ground water height is very close to the grade elevation (Refer to Subsection 2.5.4.6 and Figures 2.4-14 2.4-15 and 2.5-13)."

2. All foundations founded on the overburden have the potential to settle due to liquefaction in an SSE. If a shallow founded building adjacent to a safety-related structure settles due to liquefaction, it has the potential to damage the adjacent safety-related structure. Hence, all shallow founded structures that are adjacent to a safety-related structure will be founded on rock or all of the underlying overburden will be removed and replaced with lean concrete to minimize any potential damage to a safety-related structure due to liquefaction or demonstrate that failure of the non-safety-related structures will not damage safety-related structures.

The following statement will be added to PSAR Subsection 2.5.4.8 immediately after the first sentence: "To minimize potential damage to a safety-related structure due to the liquefaction of soils under adjacent non-safety-related structures, the adjacent non-safety-related structures will be either founded on rock or the underlying overburden will be removed and replaced with lean concrete or demonstrate that failure of the non-safety-related structure will not damage safety-related structures."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)： 02-007

問題章節(PSAR Section)： 2.3.1 Local Meteorology

初提日期(Question Date)： 1998. 04. 08

問題內容(PSAR Question)：

一、2.3.1.3.3有關鋒面造成嚴重降水之觀測欠缺圖表觀測數據，請補充。

二、2.3.2.1.4龍門廠區露點溫度之觀測紀錄自1994年3月至1997年2月，但依表2.3-88所列龍門廠區之露點觀測儀器之開始使用時間為1994年4月顯然不一致，請說明。

三、表2.3-29低塔低層風力觀測紀錄自1981年元月起，但1980年7月起便以裝置完成風力觀測儀器，何以未涵括起初半年之觀測資料？

四、表2.3-62～表2.3-87之風速觀測紀錄有些月份之觀測總時數偏低，請說明其data recovery究竟多少？是否符合R.G.1.23要求至少90%之規定？

問題答覆(Responses)：

一、補充1958年到1984年有關鋒面豪雨資料如附件：

該期間各年鋒面豪雨發生天數如圖2.3.1.3.3-1

該期間各月份鋒面豪雨發生天數如表2.3.1.3.3-2

該期間各地鋒面豪雨發生天數如表2.3.1.3.3-3

該期間台灣北部地區各級豪雨發生天數如表2.3.1.3.3-4

二、廠址低塔氣象觀測系統（包括露點溫度計）係於1994年2月底汰換安裝完成，試運轉驗收需時一個月，PSAR中為求露點溫度有完整三年資料，故記錄取自1994年3月至. 1997年2月，而表2.3-88中所列開始時間係以正式驗收後入財產帳為準，因此為1994年4月。

三、表2.3-29為低塔EL.21 M降水時之風向風速頻率資料，由於記錄已達16年以上，為擷取完整年份資料，因此割捨初始半年資料。

四、表2.3-62 ~表2.3-87為高塔1987-1996十年之大氣穩定度與風向風速聯合頻率，各月之data recovery如下：

一月	二月	三月	四月	五月	六月	七月	八月	九月	十月	十一月	十二月	全年
82.8	77.3	87.3	98.6	95.4	90.0	85.4	97.8	96.9	95.9	98.1	95.3	91.8

其中一、二、三及七月因1987、1988年1 ~ 3月間溫差儀故障及1993年7月間儀器系統曾遭雷擊，故Data recovery均未達90 %，惟全年平均仍達91.8 %。俟FSAR時，可去捨1987、1988前兩年資料統計之，則各月之Data recovery預計將可達90 % 以上。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 02-008

PSAR Sections: 2.5.4.6

Question Date: April 3, 1998

PSAR Question:

The underground water table at the site is close to the surface and if only drainage ditches and water sumps are used during construction to maintain dryness at the excavation area, it is doubtful it will prevent the same situation that happened in Chinshan unit 2 where water intrusion was found under the torus. Please explain.

Response:

The Lungmen NPS Units 1 and 2 shall have a permanent dewatering system for operation. See PSAR Subsection 2.5.4.12. In addition, a waterproofing membrane having a life time of 40 years is being implemented. Both features combined will prevent water intrusion as it occurred in Chinshan unit 2.
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: 02-009

PSAR Sections: 2.5.4.12

Question Date: March 10, 1998

PSAR Question:

After improvements to subsurface conditions are done, evaluations should be performed to determine the strength of the rock and when loads are applied, their impact on the rocks. Please explain.

Response:

The foundations of all major buildings, including safety and non-safety-related are deep in the intact rock. No foundation treatments are expected or required. If the actual bedrock conditions are not as previously assumed a treatment program will be developed, executed, and recorded. The results of the comparison of assumed rock conditions with actual and any executed treatment programs will be described in the FSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 02-010

PSAR Sections: 2.3

Question Date: March 23, 1998

PSAR Question:

1. It was discussed in PSAR Chapter 15 Accident Analysis that the Standby Gas Treatment System requires a stack height of 116 meters. Is the meteorological data sufficient to cover this stack design? Please clarify.

2. Has proper evaluations been performed for the possibility that the stack might collapse and jeopardize other safety related structures in the plant as required by R.G. 1.70 Section 2.2.1? Please explain.

Response:

1. Yes, a review of Regulatory Guide 1.23 and SRP 2.3.3 show sufficient data and measurements made at various heights consistent with the regulatory requirements..

2. SRP Section 2.2.3 does not require a failure analysis of any structure located on the Lungmen NPS Unit 1 and 2 site. PSAR Section 2.2.3 is for offsite hazard evaluations. The SGTS stack design will be documented in Subsection 3.8.4 of the FSAR. It will be designed such that it will not collapse in an SSE nor fail to perform its safety-related function.

   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: 02-011

PSAR Sections: 2.5.4.6

Question Date: March 10, 1998

PSAR Question:

This section stated that the groundwater level at Lungmen is already close to surface. What is causing it and is it still rising? What are the impacts to the site? Please explain.

Response:

The ground water in the Lungmen NPS site area is divided into two regions. A northern region and a southern region. The two regions are divided by Chihting Chi Creek. Lungmen NPS is located South of Chihting Chi Creek in the Southern region. A ground water monitoring program was put in place at Lungmen NPS site and its vicinity (See PSAR Subsection 2.4.12.1.3). Based on these observations it was determined that the fluctuations in groundwater level are seasonal in nature and are related to the amount of precipitation in the region, (See PSAR Subsection 2.4.12.2.3). Only 1 well (GM1), which is located North of Chihting Chi has records that show water levels above plant finished grade.

Since the design basis for hydrostatic loading is at plant finished grade (See PSAR Subsection 2.4.12.4), and no observation wells South of Chihting Chi Creek have records that show water levels above plant finished grade (PSAR Figure 2.4-14), the fluctuations of water level will have no impact on the site.

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

編號(Track Number)： 02-012

問題章節(PSAR Section)： 2.5.1.2, 2.5.4.2

初提日期(Question Date)： 1998.5.18

問題內容(PSAR Question)：

2.5.1.2中廠址岩石屬於Wenkzkeng Formation，而2.5.4.2中岩石屬於Tatungshan strata與Wuchishan strata，兩處使用之名詞不一致，易造成混淆，是否考慮其一致性，請澄清。

問題答覆(Responses)：

2.5.1.2中說明廠址岩石屬於Wenkzkeng Formation (or Tatungshan Formation) and Mushan Formation；由於Wenkzkeng Formation及Tatungshan Formation之岩性相類似，兩者可相互對比，因此，早期以分佈於屈尺斷層以北者稱為Wenkzkeng Formation，分佈於屈尺斷層以南者稱為Tatungshan Formation；而最近由中國地質學會於核四地質複查工作中所完成之廠區地質圖中均統稱為Tatungshan Formation，且PSAR 2.5.1.2中亦僅就Tatungshan Formation作詳細說明，因此，修改2.5.1.2第二段、第一行內容Wenkzkeng Formation (or Tatungshan Formation)為Tatungshan Formation (or Wenkzkeng Formation)，以求統一，詳如附件。

另外，Mushan Formation為上覆於Wuchishan Formation之上，兩者岩性類似，雖不易區別，但仍屬不同岩性地層單位；因此，於2.5.1.2及2.5.4.2中主要係分別依據該二地層單位名稱來編寫，應屬正確，擬不予修正。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 02-013

PSAR Section: 2.5.4.6

Question Date: May 18, 1998

PSAR Question:

1. This section stated that pumping arrangement will keep the plant site water table under control. Please confirm whether there will be pumping system in Lungmen in future.

2. Since pumping system will affect the groundwater conditions, please confirm if there is plan to investigate the groundwater usage and the impact on the groundwater distribution. This investigation should be conducted for unit 1 and unit 2 locations.

3. The groundwater distribution (such as Figure 2.4-8 to 2.4-10) should include the water flow direction and velocity.

RESPONSE:

1. As committed in PSAR, permanent dewatering systems for major buildings will be provided in order to eliminate the groundwater seepage problem.
2. The design basis for the dewatering system is that the running of the system should not result in a lowering of the water table. Since only a very localized reduction in the water table is expected, the investigation on the effect of the dewatering system on civilian water use is not conducted.
3. The aquifer coefficient of conductivity in Lungmen region is 0.0002 cm/sec (2.4.12.2.4). Because Lungmen Nuclear Power Plant does not use groundwater for operations and running of the dewatering systems does not change the flow pattern of groundwater, further discussion on flow directions and velocities of groundwater is not required in PSAR.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 02-014

PSAR Sections: 2.5.4.5

Question Date: May 18, 1998

PSAR Question:

This section on the site backfill monitoring plan only showed some figures. Is there any backfill plan and relevent control and audit procedures? Please explain.

Response:

For Lungmen NPS, the technical requirements for excavation and backfill were developed by the design engineers responsible for the facility/structure. The guidelines for the construction QC/QA program are provided in those technical requirements. In the construction plan for excavation and backfill, the technical requirements and the QC/QA program were included as a part of the overall plan to govern the quality of the field excavation and backfill work.

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

Track Number: 02-015

PSAR Sections: 2.5.4.3

Question Date: May 18, 1998

PSAR Question:

Table 2.5-10 listed the mechanical properties of the rocks at the Lungmen site. What sampling basis has been used? Is it representative of the unit 1 and unit 2 conditions? Please explain.

Response:

The mechanical properties listed in Table 2.5-10 covers a range of mechanical test analysis results for rock samples taken from Unit 1 and Unit 2 sites during the Phase II geological survey.

PSAR Table 2.5-10 will be changed as follows: The column headings will be modified from "Sample 1" and "Sample 2" to "Unit 1" and "Unit 2".

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)：　02-016

問題章節(PSAR Section)：　2.0

初提日期(Question Date)：　1998.5.18

問題內容(PSAR Question)：

有關核四廠址古河道之變遷狀況，對廠址之可能影響，宜作一補充說明。

問題答覆(Responses)：

分佈於核四廠廠區範圍內之古河道遺址，於目前一號機廠區開挖之地質剖面中清晰可見，這些古河道之自然堆積物均屬於廠區地表覆蓋層之一小部份，堆積分佈情形相當平整，並未影響廠址地質之穩定性，並且於開挖工作中將完全挖除掉。

未來FSAR所包含之開挖地質圖測繪資料，將明白表示古河道堆積分佈情形，並有完整說明。

編號(Track Number)：　02-017

問題章節(PSAR Section)：　2..5.4.2.3

初提日期(Question Date)：　1998.5.18

問題內容(PSAR Question)：

有關核四安全設備地基開挖時，將進行地質圖測繪工作，其比例尺為1/200，相關重要設備基座之地質剖面圖在FSAR中是否會作一詳細的探討，請澄清。

問題答覆(Responses)：

核四廠區開挖工作將就全部開挖斜面、垂直面及基礎平面進行比例尺1/200及1/1000之地質圖測繪，並就測繪成果進行詳細評估與探討。這些地質圖測繪成果及評估結果均將記載於FSAR中。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)：　02-018

問題章節(PSAR Section)：　Sec.2.5

初提日期(Question Date)：　1998.5.10

問題內容(PSAR Question)：

一、本節於大地工程中提及之反應器形式(1986年5月)與目前確定者不符，請修正相關的敘述，如PWR與Maanshan-type plant layout等。

二、核四初期安全分析報告有關地盤、基礎穩定及邊坡穩定性之評估資料均完成於民國七十一、二年間，其時尚未確定反應器型式，而現今反應器型式已確定，是否應重新加以評估?若認為安全無慮，則建請刪去重加評估之相關詞句。

問題答覆(Responses)：
問題一及二答覆:
PSAR Sec. 2.5.(4) &(5) 將修改如下:
Section 2.5.(4) Summary of Geotechnical Engineering
The Units 1 and 2 area of Yenliao site consisted of a soil veneer comprised of unconsolidated clay, silt and sand with minor amounts of cobbley gravel, which overlie a generally hard, well-indurated series of interbedded , fine-grained sedimentary rocks. Part of site preparation, excavation of the cut slope to the south and west of the power block area the mass excavation grade work have been conducted. The recommended foundation design parameters provided in Section 2.5.4 are based on exploration and testing conducted assuming a plant grade of 12 m above sea level. Therefore, the assumed parameters for foundation design can be evaluated and determined.
Section 2.5.(5) Conclusions
(a)~(d) 不變
(e)本段最後一句修改為"The foundation rock will be mapped in detail by an experienced onsite engineering geologist as excavation proceeds."
(f) 不變

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-001

PSAR Sections: Ch 3.7.2.1.5

Question Date: December 22, 1997

PSAR Question:

1. Section 3.7.2.1.5 talks about the dynamic analysis of nuclear power plant buildings but the description in PSAR is very little. For instance, there is coupling relationship between reactor building and reactor internals and similarly, therer are coupling relationship between building structure (main system) and associated internal structures (sub-system) in control building and turbine building respectively but no explanation was given for these building in 3.7.2.3 either. Please supplement with explanation of what factors should be considered in dynamic analysis? This should make the dynamic analysis of nuclear power plant buildings more complete.

2. In Section 3.7.2.1.5 which describes the dynamic analysis of nuclear power plant buildings, should rocking effect be considered due to horizontal motion of the ground surface?

PSAR Response:

1. Analysis of Seismic Category I (SC I)structures and the Reactor Pressure Vessel (RPV) is accomplished by using the response spectrum or time-history approach as discussed in PSAR Subsection 3.7.2.1. Subsection 3.7.2.1.1 describes the equations of Dynamic Equilibrium for Base Support Excitation. Subsection 3.7.2.1.2 shows the solution of the Equations of Motion by Modal Superposition. Subsection 3.7.2.1.3 discusses Analysis by Response Spectrum Method. These are generic approaches to the dynamic analysis. As discussed in Subsection 3.7.2.1.5, the dynamic analysis of all Seismic Category I buildings will be performed using one of the methods indicated above. Dynamic modeling for Seismic Category I buildings (i.e., RB, CB and AFB) follows the approach described in Subsection 3.7.2.3.1.

   The question on coupling between system and subsystem within a single building has also been raised in Track No. 03-002. Coupling effect between main system and subsystem is considered as described in Section 3.7.2.3.1, is applied to any SC I building (e.g., the RB, CB and AFB) and is discussed further in the response to Track No. 03-002. The control building and turbine building are separate individual buildings and there is no structural coupling relationship between them (see Subsection 3.8.4).

The details of dynamic analysis performed for the RB, CB and AFB will be provided in the FSAR.

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

2. The rocking effect due to horizontal ground motion is automatically accounted for in the soil-structure interaction (SSI) analysis using SASSI code as described in Subsection 3.7.2.4. Whenever a building response is calculated from a second step dynamic analysis, rocking effects are included as input simultaneously applied with the horizontal translational motion at the basemat.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-002

PSAR Sections: Ch 3.7.2.3

Question Date: December 24, 1997

PSAR Question:

When system-subsystem interaction is considered, the methodology described in Reference 3.7-8 was employed. And hydrodynamic mass derivation, referenced in 3.7-3, was used in the modeling of reactor pressure vessel and internal. Please explain the advantages and disadvantages of the two methods introduced in those two references and its assumptions ? And since those methods are described in periodicals, what is its applicability ? any experience it has been applied to power plants? Please explain.

PSAR Response:

The Equipment-Structure Interaction (ESI) methodology described in Reference 3.7-8 was used on WATTS BAR Nuclear Power Plant of TVA which was reviewed and accepted by the USNRC. This method is based on a dynamic substructuring method in which the equipment-structure interaction system is partitioned into the Single-Degree-of-Freedom (SDOF) system representing the equipment and the equipment support impedance representing the dynamic characteristics of the structure at the equipment support. The method is mathematically exact and can be applied to any linear system-subsystem of equipment supported on structural system. The developed response spectra are more realistic than the conventional floor response spectra in which the ESI effects are ignored.

Reference 3.7-3 for hydrodynamic mass derivation has been used in all GE BWR plants since the method was developed. In order for the determination of the system dynamic response of the reactor pressure vessel (RPV) and the internals to be realistic, the inclusion of hydrodynamic mass is mandatory to account for dynamic effects of water in the RPV.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-003

PSAR Sections: Ch 3.7.4.4

Question Date: November 21, 1997

PSAR Question:

In the judgement of OBE Exceedance, the PSAR has listed Response Spectrum and CAV verifications but according to REG GUIDE 1.166, another Instrument Operability check has to be included besides the two mentioned. Please confirm and modify if needed.

PSAR Response:

Yes, the Instrument Operability Check will be included in the judgment of the OBE Exceedance.

Subsection 3.7.4.4, first sentence will change to read "Within four hours after the earthquake; the 5% damped response spectrum, the CAV, and a check on the operability of the instrumentation for each of the three components of recorded data in the free field will be obtained and evaluated to determine if OBE is exceeded and plant shutdown is required as defined in Regulatory Guide 1.166."

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-004

PSAR Sections: Ch 3.12

Question Date: November 24, 1997

PSAR Question:

In the GE ABWR SSAR there is one section on Tunnels (including Main Steam Tunnel, Safety Related Tunnel and Miscellaneous Tunnel. etc.) but the PSAR submitted has no such section. Please provide such section or explain why it is not there.

PSAR Response:

The ABWR SSAR Section 3.12, "Tunnels", is not required by Regulatory Guide 1.70, Revision 3, and Standard Review Plans for Chapter 3, but was included in the SSAR to provide a top level summary of the safety measures incorporated into overall ABWR standard plant.

The USNRC did not require a separate evaluation of ABWR SSAR Section 3.12 information for their safety evaluation of the SSAR. For the Lungmen NPS PSAR, it was thought more appropriate to adhere to the SRP and not incorporate Section 3.12, provided equivalent information will be located elsewhere in other sections of the Lungmen NPS FSAR. ABWR SSAR Section 3.12 contains information which will be included in other sections of the Lungmen NPS FSAR.

The information in ABWR SSAR Subsection 3.12.1, Main Steam Tunnel, is located in the Lungmen NPS PSAR. Detailed analysis of these features will be included in the Lungmen NPS FSAR.

Comparison of ABWR SSAR 3.12, "Tunnels" and Lungmen NPS PSAR Section 3.4, "Water Level (Floods) Design" indicates, as shown below, that much of the SSAR contents of 3.12 pertaining to floods have been included in PSAR Section 3.4:

a. SSAR 3.12.1.1: Floors and walls to be water tight.

PSAR 3.4: Covered in Section 3.4.1.1.2.

b. SSAR 3.12.2.1 through 3.12.2.3: Mainsteam tunnels are designed to withstand the effects of high energy line breaks and to vent the resulting pressure buildup to atmosphere.
PSAR 3.4: Covered in Section 3.4.1.1.2.

c. SSAR 3.12.2.1(6): Tunnels will contain leak detection equipment and provision for water removal.
PSAR 3.4: Section 3.4.1.1.2: MSL tunnel area is instrumented.

d. SSAR 3.12.2.3: Flooding of the tunnels due to site flood conditions will be precluded by protecting the entrances of the tunnel from water entry.
PSAR 3.4 : Section 3.4.1.1.2 -a, b and c covers above.

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

ROCAEC Review Comments:

8. The steam tunnel design basis specified in SSAR 3.12.1.1 could not be found in (PSAR) 3.4.1.1.2 and 3.4.8. The description of the Steam tunnel design basis consists of 2 parts, one is on the standing weight consideration and the other one is on the consideration between the buildings. Please take into account these two considerations into the relevant PSAR sections.
9. SSAR 3.12.2.1 to 3.12.2.3 are on safety-related tunnel matters and not on descriptions of steam tunnel resistance to pipe break. The response was not to the question and please re-explain.
10. PSAR only stated that the steam tunnel has capability to withstand dynamic effects of pipe break in the reactor building and control building. Does that mean turbine building is not considered?
11. Steam tunnel goes through R/B, C/B and T/B and is an integral structure. The response classified the seismic category of the tunnel according to the building it is in and it is SC I in R/B and C/B and SC II in T/B. Since the steam tunnel is an integral structure, this division into two different categories should be reexamined.
12. Please include each of the requirements in SSAR 3.12 clearly into the various sections of PSAR. The current response is too vague that they will be included.

Additional Clarifications:

8. The PSAR section 3.4 information, which is the general flood protection design criteria, will be met for all seismic category I structures in their design. As was mentioned in a response above, the steam tunnel detailed

flooding evaluation will be included in the FSAR as part of the Reactor, Control, and Turbine Building flooding evaluations. The flooding evaluation of the structure will include the expansion joints (see Comment 11 below) between the RB and CB and the CB and TB, and will consider the standing water height due to flooding.

9. The information contained in SSAR Sections 3.12.2.1 through 3.12.2.3 will be included in a new PSAR Section 3.8.1.4.8, Safety-Related Tunnels, at this time. The SSAR information will be included as applicable to the Lungmen safety-related tunnels; most of the information regarding safety-related aspects, such as earthquake, flood (including hydrostatic head), pipe break dynamic effects, fire and environmental conditions, is applicable. It should be noted that the this new section would not be consistent with the format of Regulatory Guide 1.70 and SRP 3.8.1, because some information would be out of place in accordance with the formats. Therefore, in the FSAR, the information will be distributed appropriately.

10. The steam tunnel in the Turbine Building will be designed for a high energy line break similar to the commitments for the steam tunnel in the reactor building and control building.

11. There are two expansion joints in the main steam tunnel. One is located between the Reactor Building and Control Building and the other located between the Control Building and Turbine Building. They allow the individual buildings to move independently in an earthquake. The safety class of the tunnel will change at the expansion joints between the Control Building and Turbine Building. As noted in clarification to Comment 10 above the steam tunnel in the Turbine Building will be designed for a high energy line break similar to the commitments for the steam tunnel in reactor building and the control building.

12. The PSAR Sections 3.4 and 3.8 will be revised as attachment in the PSAR amendment.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

TRACK No. 03-004

Lungmen NPS PSAR (Insertion of New Sections)

PSAR Section 3.8

3.8.4.1.8 Safety - Related Tunnels

3.8.4.1.8.1 Main Steam Tunnel

The Main Steam (MS) tunnel is a low leakage reinforced concrete structure. The MS tunnel is located well above the grade and will not be subject to inundation from site flooding. It will be designed for loads resulting from high energy pipe break inside the tunnel. In addition, the MS tunnel is designed for the loads and load combinations applicable to RB and CB for the portion inside these buildings.

There are two expansion joints in the MS tunnel. One is located between the Reactor Building and Control Building and the other is located between the Control Building and Turbine Building. They allow the individual buildings to move independently in an earthquake. The safety class of the tunnel will change at the expansion joints between the Control Building and the Turbine Building. The steam tunnel in the Turbine Building will also be designed for a high energy line break.

3.8.4.1.8.2 Reactor Service Water System Tunnel

3.8.4.1.8.2.1 Description

The purpose of the Reactor Service Water (RSW) tunnel is to provide protected and divisionalized pathway for piping, power cable and instrumentation and control cable between the Control Building and the RSW System Pump House. The RSW tunnel is a reinforced concrete structure.

3.8.4.1.8.2.2 Design Basis

(1) The tunnel will be designed to applicable safety requirements of seismic, flood and environmental conditions in order to maintain the RSW system safety function. Seismic requirements shall be in accordance with Subsection 3.7.3.12.
(2) The tunnel will be designed to ensure that the integrity of the RSW System piping penetration at the Control Building is maintained under pipe break conditions.
(3) The tunnel will be designed to ensure its design basis functions even under internal piping system breaks.
(4) Site flooding of the tunnel will be precluded by tunnel entrance protection at the pump house.

The loads, loading combinations will be in accordance with PSAR Subsection 3.8.4. The analysis results will be supplied with FSAR.

PSAR Section 3.4

3.4.1.1.2.8 Evaluation of Reactor Service Water System Tunnel

Analysis of potential flooding within the service water tunnel will be considered and the analysis results will be supplied with the FSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-005

PSAR Sections: Ch 3.13

Question Date: November 24, 1997

PSAR Question:

In the GE ABWR SSAR there is a section on Secondary Containment and Divisional Separation Zones Barrier Considerations but the PSAR submitted has no such information. Please provide this information or explain why it is not there.

PSAR Response:

The ABWR SSAR Section 3.13, "Secondary Containment and Divisional Separation Zones -- Barrier Considerations", was not required by Regulatory Guide 1.70, Revision 3, and Standard Review Plans for Chapter 3, but was included in the SSAR to provide a list of references to the rest of the SSAR and a top level summary for the convenience of the U.S. Advisory Committee on Reactor Safeguards (ACRS) of the safety measures incorporated into overall plant relative to the barrier provisions.

The USNRC did not require a separate evaluation of Section 3.13 information for their safety evaluation (FSER) of the SSAR. For the Lungmen NPS PSAR, it was thought more appropriate to adhere to the SRP, which does not include Section 3.13, provided equivalent information will be located elsewhere in other sections of the Lungmen NPS PSAR and FSAR.

The key SSAR subsection 3.13.3, General ABWR Containment Structures, Systems and Barrier Descriptions, provides references to other locations in the SSAR that deal with various information on the barriers provided in the ABWR design. The PSAR also includes counterpart locations which provide similar information as listed below:

Overall Plant Design and equipment Layout	Refer to Figures 1.2-2 thru 1.2-31
Reactor Building (RB)	Refer to Section 3.8 and Figures 1.2-4 thru 1.2-12
Secondary Containment (SC)	Refer to Section 6.2 and Figures 1.2-4 thru 1.2-12
Divisional Separation Zones (DSZ)	Refer to Figures 1.2-2 thru 1.2-12
Design Basis Accident Inside	Refer to Section 6.3 and Chapter 15

Containment

Design Basis Breaks Outside Containment Inside Secondary Containment	Refer to Section 6.3 and Chapter 15
Reactor Building/Secondary Containment/Divisional Separation Zone – HVAC Systems	Refer to Section 9.4, Figures 9.4-3 thru 9.4-5
Secondary Containment Penetrations	Refer to Section 6.2, Table 6.2-9
RB/SC Fire Hazard Analysis	Refer to Appendix 9A, Table 9A.6-2
RB/SC Flooding Analysis (Internal and External)	Refer to Section 3.4, Table 3.4-1 and Section 2.4.10
RB/SC/DSZ Safe Shutdown Equipment – Qualifications	Refer to Appendix 3I and Tables 3.6-1 and 3.6-2
Engineered Safety Features	Refer to Sections 6.2, 6.3 and 6.4
Postulated Pipe Break Aspects	Refer to Section 3.6
PRA - Plant and Public Risk Analysis	Refer to Sections 15.6.6 and 19.6 and Appendix A, Attachments AJ and AM
Technical Specifications- Containment Structures and Systems	Refer to Chapter 16

Additional detailed design information regarding barriers, which is being developed now, 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: 03-006

PSAR Sections: Ch 3A

Question Date: November 21, 1997

PSAR Question:

1. This Appendix is on Soil Structure Interaction Analysis. There is no information provided in this section. Please explain why until FSAR will there be any information provided.

2. In general, once the site is determined the Site-Specific Geotechnical Data will be collected and analyzed. Please provide the soil dynamic (non-linear) characteristics, i.e., strain dependent soil properties (such as damping, shear modulus and crevice water pressure), soil layer structures and properties (including pressure wave velocity and shear wave velocity), etc. for the SSI analysis.

3. Please briefly describe the methods for analyzing the soil structure interactions and the software used SASSI.

PSAR Response:

1. Information on SSI analysis was not provided in PSAR since the design of Nuclear Island Category I Buildings is still in progress. The results of the analysis will be supplied with FSAR. The SSI methodology is described in Subsection 3.7.2.4 of the PSAR.

2. The various soil characteristics are discussed in detail in PSAR Section 2. Strain dependent soil damping ratios and the shear modulus are shown in Figure 2.5- 26. The values of these properties are also listed in Tables 2.5-16 and 2.5-17. Soil/ Rock properties are discussed in Subsection 2.5.4.2.3. Pressure wave velocity and shear wave velocity values are shown in Subsection 2.5.4.2.4.

3. Subsection 3C.5 describes the Soil Structure Interaction Analysis Software, called SASSI. Also Subsection 3C.6 describes Free-Field Site Response Analysis Code, -SHAKE, which is used to generate input data for SASSI program. These two, along with Subsection 3.7.2.4, describe the methodology of soil structure interaction.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-007

PSAR Sections: Ch 3C

Question Date: November 21, 1997

PSAR Question:

Most of the computer programs described in this Appendix did not mention which of the four buildings listed in 3C.1 it applies to ? Please provide this information in PSAR (similar to the Extent of Application in the GE Standard SAR).

PSAR Response:

The following is the response to the question.

(1) ANSYS : This is a general purpose program and is used for the analysis of Containment Internal Structures.

(2) SSDP-2D: This program is used for Reactor Building, including RCCV, Control Building and Auxiliary Fuel Building concrete section analysis.

(3) NASTRAN: This program is used in the analysis of Reactor Building (RB) , Control Building (CB), and Auxiliary Fuel Building (AFB).

(4) SASSI: This program is used for the Soil-Structure analysis of Reactor Building, Control Building, and Auxiliary Fuel Building.

(5) SHAKE: This program is used with SASSI analysis for RB, CB, and AFB.

(6) GT STRUDL: This is a general purpose program and is extensively used on the project on various buildings such as Turbine Building, Water Treatment Building, etc.

(7) TEMCOM: This program is used for Reactor Building and Control Building thermal analysis.

(8) SCOTH: This program is used for generation of artificial time histories for seismic design ground motions for Seismic Category I Structures, Systems, and Components.

The PSAR will be revised to reflect the response above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-008

PSAR Sections: Ch 3M.2

Question Date: November 25, 1997

PSAR Question:

The URS design pressure for the low pressure piping is 0.4 x 7.07 Mpag. Please explain why the 0.4 was adopted ?

PSAR Response:

This value of 0.4 is the ratio of the low pressure piping design pressure (for ISLOCA) to the reactor coolant system (RCS) normal operating pressure. Pipe designed for 0.4 times RCS operating pressure will maintain its integrity if exposed to RCS operating pressure. Using a design pressure of 0.4 times the normal RCS operating pressure and schedule 40 pipe provides basis for assuring the pipe can withstand full reactor pressure without bursting. This value was proposed by GE and adopted by NRC for the certified ABWR (Reference: NUREG -1503, ABWR FSER Section 3.9.3.1.1)

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-009

PSAR Sections: Ch 3MA.11.1.2

Question Date: November 25, 1997

PSAR Question:

It was mentioned that "CRD piping is not upgraded to the URS design pressure because the maximum static head is 0.159 Mpag". Please explain why the static head and design pressure have nothing to do with URS.

PSAR Response:

This referenced sentence is not clear as to which system pipe it addresses and shall be deleted. The sentence applies to the CSTF portion of the CRD suction line which is not exposed to the RCS pressure and is covered by Section 3MA.11.1.1. This piping section is an extension of the low pressure sink and does not need to be designed to the URS design pressure for RCS operating pressure per Lungmen PSAR Section 3M.3 item (3). The CRD system suction piping identified in 3MA11.1.2, second item, is designed to URS design pressure as described in Section 3MA.5.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-010

PSAR Sections: Ch 3.6.2.1

Question Date: January 2, 1998

PSAR Question:

This section describes the criteria used to define break and crack locations. Since in high-energy fluid system only pipe break is considered and in moderate-energy fluid system, crack can be considered. So it is essential to define clearly the pipings as either high-energy fluid system or moderate-energy fluid system. In Section 3.6.2.1.1, piping systems are classified as moderate-energy fluid systems when they operate as high-energy piping for only short operational periods. By this classification, HPCF, RCIC and SLC will be classified as moderate-energy fluid systems. But in SRP classification no such exception was adopted. There, when break or crack was to be decided, crack was considered for high-energy fluid system which operates for only short periods. It is questionable that in the PSAR such change in the definition was adopted for the exceptions. Please clarify.

PSAR Response:

The definition of moderate-energy fluid systems provided in Section 3.6.2.1.2 is in accordance with SRP 3.6.2, Revision 1 - July 1981. Page 3.6.2-16 of SRP 3.6.2 states that, through-wall leakage cracks instead of breaks may be postulated in the piping of those fluid systems that qualify as high-energy fluid systems for only short operational periods but qualify as moderate-energy fluid systems for the major operational period. Note 6, at the bottom of page 3.6.2-16 of SRP 3.6.2 states that, an operational period is considered "short" if the fraction of time that the system operates within the pressure-temperature conditions specified for high- energy fluid systems is about 2 % of the time that the system operates as a moderate-energy fluid system.

The HPCF and SLC fluid systems inside containment are defined as high-energy fluid systems. The HPCF fluid system outside containment up to the outermost isolation valve is defined as a high-energy fluid system. The HPCF fluid system outside containment beyond the outermost isolation valve is defined as a moderate- energy fluid system, since it operates at high-energy fluid temperature and pressures about 2% of the time that the system operates as a moderate-energy fluid system. The SLC fluid system outside containment up to the injection valves is defined as a high-energy fluid system. The SLC fluid system outside containment beyond the injection valves is defined as a moderate-energy fluid system, since it operates at high-energy fluid temperature and pressures about 2% of the time that the system operates as a moderate-energy fluid system.

The steam supply portion of the RCIC system up to the RCIC turbine is defined as a high-energy fluid system. The remainder of the RCIC system is defined as a moderate-energy fluid system, since it operates at high-energy fluid temperature and pressures about 2% of the time that the system operates as a moderate-energy fluid system.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-011

PSAR Sections: Ch 3.6.1.1.3

Question Date: January 2, 1998

PSAR Question:

In the 3.6.1.1.3 section on Assumptions, failure of non-Seismic Category I piping will not have the same effect as essential system, component or equipment. But in section 3.2, certain non-Seismic Category I steam line was defined as category II considering seismic dynamic effects. Should this seismic dynamic effects be used to classify a system as essential system ?

PSAR Response:

No. The fact that the non-Seismic Category I main steam line described in Section 3.2 is dynamically analyzed for seismic loads does not imply that portion of the system should be classified as an essential system.

Section 3.6.1.1.3, item 5, states that the failure of a non-Seismic Category I piping must not result in failure of essential systems, components and equipment. In order to satisfy this assumption, the essential systems, components and equipment will be protected from any postulated failures of non-Seismic Category I piping, such as the non-Seismic Category I steam line. The non-Seismic Category I steam line is dynamically analyzed to ensure that it will not fail during a seismic event.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-012

PSAR Sections: Ch 3.6.1.3.2.3

Question Date: January 2, 1998

PSAR Question:

It was explained in this section that the combination of both SSE and HELB is considered in the design of steam tunnel so that steam tunnel will maintain its integrity under the conservative load. Therefore, HELB in the steam tunnel will not affect control room habitability. But has the dynamic effect accompanying HELB been considered ? There are some high energy pipings in the steam tunnel that are not essential systems. Has their effect on control room habitability been evaluated ?

PSAR Response:

Yes, the dynamic, flooding, and pipe reaction effects of HELB in the steam tunnel are considered in the structural design of the Main Steam tunnel. The pressure and dynamic loads including pipe reaction effects are specified as the a" and the " loads in Section 3.8.1.3.5. The structural design requirements for the steam tunnel for these loads is documented in Section 3.8.4.

Yes, the effect of HELB beyond the outboard isolation valve (in essential and non-essential piping) on control room habitability is evaluated. Since, the steam tunnel is vented to the Turbine building as described in Section 6.2.3.2, a HELB inside the steam tunnel will not effect control room habitability.

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

ROCAEC Review Comments:

Is there any non-SC I pipings in steam tunnel ? Is steam tunnel SC I structure ? If yes, then under SSE, would those pipe breaks cause damage to the control room ?

Further Clarifications:

Yes, the following non-SC I pipe lines are in the steam tunnel: The sections of main steam piping and feedwater piping from the seismic guides to the exit points from the tunnel toward the turbine building (see PSAR Figures 3.2-1 and 3.2-2), and a section of the main steam drain line from the second isolation valve to the exit point from the tunnel toward the turbine building (see PSAR

Figures 3.2-1), and a section of the RCIC steam drain line up to the exit point from the tunnel toward the turbine building.

Pipe breaks are postulated for all high-energy piping in accordance with the criteria in PSAR Sections 3.6.2.1.4.4 and 3.6.2.1.4.5. High energy piping include SC I as well as SC II piping that have maximum operating temperature >93° C and maximum operating pressure >1902.5 kPaG (see PSAR Section 3.6.2.1.1). Pipe whip restraints (PWRs) are provided, if necessary, for all high-energy lines in the steam tunnel in accordance with PSAR Sections 3.6.1.3.2.4 and 3.6.1.3.3. The PWRs would be provided to protect the steam tunnel from a potential impact from pipe whip in accordance with PSAR Sections 3.6.2.3.2. Furthermore, the steam tunnel is also designed for pressurization resulting from postulated high-energy line breaks (see PSAR Sections 3.6.1.2 and 3.8.4).

The steam tunnel, which is a portion of the control building, is a SC I structure (see PSAR Section 3.8.4, Page 3.8-29, and Section 3.8.4.1.2).

As discussed above, the control room is protected from high-energy line breaks in SC I as well as SC II piping.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-013

PSAR Sections: Ch 3.6.2.1.4.2

Question Date: January 2, 1998

PSAR Question:

1. In the description of ASME Code Section III, Class 2 Piping, Level A and Level B stress limits are specified in the system's Design Specification. Has OBE event been considered?

2. "(6) Sleeves...". Should the "Sleeves" be "Guard pipe" instead ?

3. "(7) A 100% volumetric inservice examination of all circumferential and longitudinal pipe welds would be conducted...". Why it is limited to "all circumferential and longitudinal pipe welds" but not "all pipe welds"? Please explain.

PSAR Response:

1. Yes, OBE has been specified as a service level B load. OBE is considered to be an occasional load.

2. No, "Sleeves" is correct.

3. The 100% volumetric inservice examination should be conducted for all pipe welds.

PSAR change is required. PSAR Section 3.6.2.1.4.2 item (7) will be revised to specify "all pipe welds" instead of "al circumferential and longitudinal pipe welds".

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

ROCAEC Review Comments:

2. It is obvious that sleeve is not guard pipe and in SRP 3.6.2, guard pipe was defined but not sleeve. Is it feasible to apply the conditions used on guard pipe to sleeve ? Please explain the differences between the two.

Further Clarifications:

2. Reference to guard pipes in Section 3.6.2.1.4.2 item (6) was deleted starting with the SSAR because as noted in SSAR Section 3.6.2.4 the ABWR containment design does not require guard pipes. Similarly, the Lungmen NPS containment design does not require guard pipes, and reference to guard pipes has also been deleted from Section 3.6.2.1.4.2 item (6) in the Lungmen NPS PSAR.

   The guard pipe requirement is unique to the Mark III design for BWR 6 plants. In the Mark III design, there is an annulus between the drywell wall and the containment wall. If a section of high energy process piping in the annulus were not isolatable, the NRC has required that a guard pipe be provided to discharge the liquid from a postulated break in that section back to the drywell so that the discharge does not bypass the wetwell and pressurize the containment. In the ABWR containment design, the drywell wall is the containment wall, and, therefore, no special safeguard is required as is the case in other plants with Mark I and Mark II containment design.

   As noted on page 3-34 of the FSER, the NRC staff viewed the SRP 3.6.2 definition of the term "Guard pipe" in a broad context to include "Applicable sleeves in the containment penetration area." Accordingly, the term sleeves was substituted in Section 3.6.2.1.4.2 (item 6) in the SSAR for cases in which sleeves may be required to be used in the broad context of guard pipes.*
   With respect to sleeves, the ABWR containment design is most similar to the Mark II containment design which includes sleeves only for the purpose of connecting high temperature piping to the containment. For example, the Limerick NPS, which has a Mark II containment, contains containment penetration sleeves but does not have any guard pipes. Also, Chinshan NPS, which has a Mark I containment, contains containment penetration sleeves but does not have any guard pipes. As an extension of the containment, these sleeves are designed to the pressure and temperature of the containment.

   Sleeves used in Lungmen to connect high temperature piping to the containment are designed consistent with the criteria presented in Section 3.6.2.1.4.2 (item 6), however another section is controlling regarding the design pressure and temperature for penetration sleeves. Subsection 3.6.2.1.4.2 (item 6)(a) differs with the design pressure and temperature

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* *A clarification will be added to the PSAR Section 3.6.2 to note that Section 3.6.2.1.4.2 (item 6) is applicable to sleeves required to be used in the broad context of guard pipes. However, no such sleeves exist in the Lungmen NPS design. All sleeves in the Lungmen design which function solely for connecting high temperature piping to the containment will be designed to the pressure and temperature criteria provided in Section 3.8.2.*

defined in Section 3.8.2 (Steel Components of the Reinforced Concrete Containment). As Section 3.8.2 specifically addresses containment penetrations, it is controlling for penetrations, and Lungmen sleeves are designed to the pressure and temperature therein defined. Since the sleeves are part of the containment, they are designed to the same pressure and temperature as the containment. (Similar design criteria were used in the design of Limerick.)

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-014

PSAR Sections: Ch 3.6.2.1.4.3

Question Date: January 2, 1998

PSAR Question:

1. In Item (2), does the Eq. (10) belong to NB-3653 ?
2. In Item (2), the intermediate location was identified not only by Eq. (10) but also Eq. (12) and (13). Whereas the SRP only uses Eq. (10). Please explain.

PSAR Response:

1. Yes, Eq. (10) does belong to NB-3653.
2. Section 3.6.1.1.1 identifies three exceptions taken to the NRC Branch Technical Position, MEB 3-1, included in SRP 3.6.2. These exceptions were agreed to by the NRC. The NRC agreed to the third exception because MEB 3-1, B.1.C(1)(b) had inadvertently not included the requirement that along with Eq. (10), Eq. (12) and (13) also had to exceed 2.4 Sm in order for an intermediate break to be postulated. Therefore, Section 3.6.2.1.4.3 item (2), is correct and consistent with SRP 3.6.2 and MEB 3-1, as modified by the exception to MEB 3-1, B.1.C(1)(b) stated in Section 3.6.1.1.1 (3).

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-015

PSAR Sections: Table 3.2-1b & c

Question Date: February 23, 1998

PSAR Question:

1. PSAR classified the Seismic Category for those not in Category I as Category IIA, IIB and IIC but in Table 3.2-1b & c on the classification of Structures, Systems and Components, those classified as Category II were not further classified (into A, B and C) and only indicated they will be classified in FSAR. Please explain.

2. Since PSAR already defined non-seismic category I as seismic category II but in Figure 3.2-1 the term non-seismic category I was still used which is inconsistent with the definition in PSAR. Please use a consistent definition in the figure to minimize confusion.

3. PSAR classified the safety related SSCs as Seismic Category I but the following safety related SSCs were still classified as Seismic Category II :

(1) Pipe whip restraint of Main steam system
(2) Cables of LDI
(3) Instrument racks- mechanical and electrical with safety related functions of LCPR
(4) Pipe whip restraint of feedwater line
(5) Safety-related electrical wiring penetrations of Raceway system
(6) Foundation work of Reactor Building

Response:

1. The detailed classification of structures, systems and components into seismic subcategories IIA, IIB and IIC is currently in process and was not available at the time of writing the PSAR. This classification requires consideration of plant arrangement details and component functions. These subcategory classifications will be provided in the FSAR.

2. Agreed. Figure 3.2-1 will be updated. All occurrences of non-seismic category I will be changed to seismic category II.

3. The responses are as follows:

(1) System B21 (MS) (Item 6): Pipe whip restraints (PWRs) are not part of the reactor coolant pressure boundary. The function of a PWR is to restrain the pipe in the event of a pipe break such that the broken pipe

does not adversely affect safety-related components. Thus, PWRs are classified as Seismic Category II not Seismic Category I and no change is required to Table 3.2-1.

(2) System C73 (LDI) (Item 1): Change Seismic Category to I/II
(3) System H21 (LCPR) (Item 5): Change Seismic Category to I
(4) System N22 (FW) (Item 4): Same response as for (1)
(5) System R51(RCWY) (Item 2): Change Seismic Category to I/II and also Change Safety Class to 2/N. The safety related electrical penetrations (EPENs) are a part of PCV, and therefore, should correctly be Safety Class 2.
(6) System U71 (RB) (Item 5): The foundation work of Reactor Building classified as Seismic Category II deals with excavation / backfilling / dewatering only. The PSAR will be revised by including "(Excavation/Backfilling/Dewatering only)" under "Foundation Work" as a second line of Item 5.

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

ROCAEC Review Comments:

1. TPC has responded to the original question and claimed that the categorization of seismic subcategories IIA, IIB and IIC can only be performed after consideration of plant arrangement details and component functions so it will be deferred until FSAR. But Table 3.2 has already provided complete description of SC I and II for the various systems and components and why the subcategorization can not be performed ? And the definitions of IIA, IIB and IIC are very clear so the subcategorization should be able to be completed during PSAR. Also, dynamic analysis has to be performed for some IIAs and not others, so the design flexibility will be too much if subcategorization is not done now and deferred till FSAR.

2. Please provide modified figure.

3.(1) Even though pipe whip restraints are not RCPB equipment but its failure could cause damage to SC I equipment so it should still be classified as SC I.

(4) Same reason as (1) above

Further Clarifications:

1. Whereas the definitions of seismic categories IIA, IIB and IIC are very clear, it is the assignment of SSCs to IIA and IIC that requires detailed

study of plant arrangement. (On the other hand, the assignment of equipment to IIB would be relatively straight forward, because its definition is based on specific function of SSCs, i.e., radwaste management.) The SSCs that are classified as SC II, are required to be studied for their potential collapse and jeopardy to the other safety related SSCs in the vicinity. SC IIA have high potential and IIC have negligible potential to affect other safety related SSCs. To expedite the assignment of IIA and IIC at this time would require a judgment subject to verification later when plant arrangement details are available.

Since the task of developing assignments to IIA, IIB and IIC would be continuing after the ROC-AEC review for the CP is completed, it was decided that these details can be reliably presented in the FSAR. Recognizing the AEC desire for assignment of Category II SSCs into IIA, IIB and IIC earlier in the project, the assignment process will be accelerated and documented in the PSAR Amendment.

2. As mentioned in the previous response, Figure 3.2-1 will be updated. All occurrences of non-seismic category I will be changed to seismic category I.

3.(1) As stated in PSAR Section 3.2.1, pipe whip restraints (PWRs) are safety-related components which need not function during but shall remain functional after the event of an SSE. During an SSE, the main steam and feedwater piping PWRs remain free, which means they are not required to engage and support pipe, and they are not required to perform any SC I safety function, such as maintaining the pipe integrity during an SSE. Therefore, during an SSE, the integrity of these PWRs themselves is not challenged and they are not required to perform any safety related function that require them to be classified as SC I. These PWRs are basically hinged rods (see PSAR Figure 3.6-2), which would experience insignificant vibration due to its own mass inertia only and cannot break during an SSE as the question suggests; and thus, these PWRs remain functional during an SSE. The sole function of these PWRs is to engage and restraint only a broken main steam or feedwater pipe from whipping around and damaging nearby safety related structure or equipment. If a pipe break is assumed to occur during an SSE, the PWR will perform its primary safety related function of restraining the pipe to prevent any damage, but it would not be performing (per definition of SC I requirements) any safety related function of maintaining pipe integrity, which is already assumed lost.

(4) Same as response for (1).

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

ROCAEC Review Comment 8/24/98, transmitted by SWT-TPC-001446, 8/28/98:

1. TPC committed that the Lungmen SC II SSCs will be further categorized as IIA (dynamic analysis), IIA, IIB and IIC and will be completed in PSAR stage. Please furnish the categorization as soon as possible for the CP issuance.

Further Clarification:

1. Lungmen SC II SSCs have been categorized as IIA, IIB and IIC as shown in attached Table 3.2-1b, Table 3.2-1c , Figure 3.2.1 and Figure 3.2.2

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-016

PSAR Sections: 3.2.5.3

Question Date: February 23, 1998

PSAR Question:

1. It is mentioned in note f of Table 3.2-1d on Seismic Category IIA that SSCs will be analytically checked to determine that they will not collapse when subjected to SSE load. Please clarify if the meaning of "analytically check" and "collapse" are consistent with the requirements of "dynamic analysis" and "structural integrity" defined in Section 3.2.5.3.

2. Based on the requirements of Section 3.2.5.3 of PSAR, the main steamline in the turbine building is classified as seismic category II. And the portion that provides main steamline leakage path will be analyzed using a dynamic seismic analysis method to demonstrate its structural integrity under SSE loading conditions. Please clarify if this portion is classified as SC IIA.

3. To provide input load value to the main steamline dynamic analysis, turbine building should be dynamically analyzed. Please clarify if this part is classified as SC IIA.

4. The main steamline from the seismic guide up to but not including the turbine stop valve (including branch lines to the first normally closed valve) is classified as QG B. QG B should be safety class 2 but in Table 3.2-1 it was classified as non-safety class which does not conform to R.G. 1.26. Please clarify.

Response:

1. Note f of Table 3.2-1d is a general note applicable to all Seismic Category IIA structures, systems, and components (SSCs). Analytic check does not necessarily mean dynamic analysis. "structural integrity" and "to collapse" as stated in footnote f are synonymous as the stresses beyond yield are allowed but the strains are kept within acceptable ductility limits for Category IIA SSCs.

   Section 3.2.5.3 applies to the main steamlines (MSLs) only; the structural integrity of the MSLs will be demonstrated by a dynamic seismic analysis. There is no change required to PSAR from the above response.

2. Seismic classification of structure, systems and equipment in the Lungmen PSAR are defined as either Seismic Category I, II or non-seismic. As stated in Note of PSAR Table 3.2-1d, the breakdown of Seismic Category II items into IIA, IIB and IIC will be done in the FSAR.

   When the FSAR is prepared, the main steam lines in the Turbine Building will be classified as Seismic Category IIA based on the requirement of PSAR Section 3.2.5.3 that the main steam leakage path is used to mitigate the consequences of an accident and must remain functional during and after an SSE or OBE.

3. Dynamic analysis is performed for the Turbine Building to generate Amplified Response Spectra (ARS) for the SSE and OBE conditions for use in design of the main steam lines. Refer to Response to Item 2) for classification of the main steam lines.

4. Safety classifications are assigned based on the function of each structure, system and component. As stated in Section 3.2.3 of the Lungmen PSAR, ANS Standard 52.1 is used to assign safety classes to these functions. The function of the main steam lines from the seismic guide up to, but not including, the turbine stop valves is to hold up and allow for plateout of the fission products following an accident. This function is not safety related as stated in PSAR Section 3.2.5.3 and PSAR Table 3.2-1c, Item B21 (1).

   Due to the importance of this function; however, the pertinent requirements of Appendix B, 10CFR Part 50 have been applied even though not strictly required by its non-safety classification. Therefore, it has a Quality Group B and non-safety classification.

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

ROC-AEC Review Comments:

1. The clarifications on analytic check involve not only dynamic analysis but non-dynamic analysis as well. What is the guideline for distinguishing the two ? Dynamic analysis was considered for components of SC IIA, and what is the difference between this and for SC I ?

2. The main steam line, which was classified as SC II, should consider dynamic analysis when subcategorization is performed.

3. Turbine building should consider dynamic analysis when subcategorization is performed.

4. The section between the outboard isolation valve and the turbine stop valve (not including stop valve) is categorized as QG B seismic category I at the 1st and 2nd NPP. Furthermore, the main steam line is not used for leakage path. Now at Lungmen, the section between seismic guide and turbine stop valve is seismic IIA, dynamic analysis is performed and also used for leakage path. But why the downgrade ? This is quite unreasonable and please provide further explanation. Here the key is the seismic guide. What is the definition of a seismic guide ? Is it similar to the interface restraint in SSAR ?

Further Clarification:

1. As stated in Section 3.7, in accordance with RG 1.29 all Lungmen Structures, Systems and Components (SSC) are defined as either Seismic Category I, Seismic Category IIA, Seismic Category IIB or Seismic Category IIC. Methods of seismic subsystem analysis are described in Section 3.7.3. In determining response to seismic loading conditions, static analyses may be employed for simple, single degree of freedom subsystems, as described in Section 3.7.3.8. Dynamic analyses are generally employed for multiple degree of freedom subsystems which are mathematically more complicated and either could not be adequately characterized for static analysis, or would require excessive levels of conservatism in their characterization to allow for static analysis. Thus, a relatively simple Seismic Category I component, such as a floor mounted safety related pump, may be analytically qualified for seismically induced loads by static analysis, while a multiply supported non safety related Seismic Category II subsystem, like main steam lines, is more appropriately dynamically analyzed for seismic loading. The primary difference in qualification between Seismic Category I and Seismic Category II lies in the acceptance criteria to be met under seismic loading, as defined in Section 3.2.

2. & 3.

The response to ROC AEC review comment number 1 above indicates the main steam line will be dynamically analyzed as suggested. The details and results of this analysis will be available at the FSAR stage.

4. The Lungmen NPS Unit 1&2 seismic design, as described in PSAR section 3.2.5.3, is consistent with the original design as described in the GE Advanced BWR (ABWR) Standard Safety Analysis Report (SSAR) section 3.2.5.3 and in the US NRC Design Certification Safety Evaluation Report (NUREG-1503) for the ABWR design. Thus, the design as described is consistent with the design licensed by the US NRC.

Terminology changes from seismic interface restraint (ABWR) to seismic guide (Lungmen), and from non-Seismic Category I (ABWR) to Seismic Category II (Lungmen) exist.

In this way, seismic-generated motion for the MS line is restrained at the RB outboard isolation valve and at the turbine stop valve and is limited in the interface area without direct restraint.

The "seismic guide" referred to in PSAR Section 3.2.5.3 is same as the "seismic interface restraint" in SSAR Section 3.2.5.3. The term "guide" is more appropriate than "restraint," since the pipe is guided to move freely except laterally and vertically.

Further Comments from AEC Received 8/24/98:

The main steam line between seismic guide and stop valve is used as leakage path in Lungmen and should be classified as SC I according to R.G. 1.29. Even though SRP stated that in BWR-6 (not used for leakage path) this portion can be classified not as SC I but justification has to be provided. One should not claim one can so classify the main steam line simply because SSAR said so. Please provide further explanations. Also, at Kuosheng, this section of the pipe was classified as SC I, QGB, even not used for leakage path, which implies Lungmen has downgraded this section of the pipe and safety standard. Please restore the pipe classification of leakage path piping back to the same classification to that of Kuosheng to satisfy the requirements of R.G. 1.29.

Further Clarification:

In order to clarify the issue of the classification of the seismic category of the main steamline between the seismic guide and the turbine stop valve, three important documents should be referred to in which the USNRC provided their approval to change the seismic classification of the main steam piping downstream of the seismic guide from seismic Category I to non-seismic Category I. In addition, some discussion on the justification for the seismic classification change is also provided below.

The seismic guide is a two way restraint that will provide a structural barrier between the seismic Category I portion of the main steamline in the Reactor Building / Control Building steam tunnel and the non-seismic Category I portion of the main steamline in the Turbine Building. The seismic guides are located adjacent to the Nuclear Island (NI) / Turbine Island (TI) interfaces.

The following three USNRC documents show USNRC approval and basis of approval of the classification change from Seismic Category I to Non-Seismic Category I:

(1) SECY-93-087, Dated April 2, 1993, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs.":

Section II.E of SECY-93-087 discussed the NRC staff's position on the portion of the ABWR design that eliminates the main steam isolation valve leakage control system. It also provides the approval for the use of an alternative leakage path that takes advantage of the ability of the large volume and surface area in the main steam piping of the ABWR, the main steam drain lines, the turbine bypass line, and the condenser to hold up and plate out the release of fission products following core damage. In this manner, the alternative leakage path and condenser are used to mitigate the consequences of an accident and are required to remain functional during and after a SSE event.

(2) NUREG-1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," July 1994.:

NUREG-1503 was issued by the U.S. Nuclear Regulatory Commission (NRC) staff in order to document the NRC staff's review of the U.S. ABWR design. The U.S. ABWR design was submitted by GE Nuclear Energy (GE) in accordance with the procedures of Subpart B to Part 52 of Title 10 of the Code of Federal Regulations.

(3) NUREG-1503, Supplement 1, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," July 1994.

Supplement 1 to NUREG-1503 documents the NRC staff's review of the changes to the U.S. ABWR design documentation since the issuance of the Final Safety Evaluation Report (FSER). GE made these changes primarily as a result of detailed engineering completed after NUREG-1503 was issued and as a result of the design certification rulemaking for the ABWR design. (Supplement 1 did not need to discuss the issue any further.) On the basis of its evaluation, the NRC staff concluded that the confirmatory issues in NUREG-1503 were resolved, that the changes to the ABWR design documentation were acceptable, and that GE's application for design certification meets the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the U.S. ABWR design.

The three documents listed above all contain words equivalent to the following: "The Main Steamline (MSL) from the seismic interface restraint up to but not including the turbine stop valve will be classified as Quality Group B and inspected in accordance with the applicable portions of ASME Section XI. This portion of the piping maybe non-seismic Category I if it has been analyzed using a dynamic seismic analysis method to demonstrate its structural integrity under SSE loading conditions. However, all pertinent QA requirements of Appendix B to 10CFR Part 50 are applicable to ensure that the quality of piping material is commensurate with its importance to safety during normal operational, transient and accident conditions."

The justification for the change of classification of the MSL from the seismic interface restraint up to but not including the turbine stop valve from Seismic Category I to non-Seismic Category I is described in the three NRC documents noted above. This justification is based on:

(a) The change of classification is needed for consistency with the non-Seismic Category I classification of the Turbine Building.
(b) The piping will be analyzed to show structural integrity under SSE loading conditions. (Note: The analytical methods and procedures will be the same as used for Seismic Category I piping. Structural integrity will be demonstrated by showing Service Level D limits will not be exceeded.)
(c) All pertinent QA requirements of Appendix B to 10 CFR Part 50 will be applied. (Note: This will be done by meeting the requirements of Quality Group B,)

In conclusion, based on the above discussion, the Lungmen main steamline from the seismic guide to the turbine stop valve can be classified as non-seismic Category I.

Safety Class 1 (SC-1) applies to all components of the reactor coolant pressure boundary (RCPB) (as defined in 10 CFR Part 50.2). These components shall be designed to remain functional in all events. Safety Class 2 (SC-2) plus dynamic analysis applies to pressure-retaining portions of the primary containment and these components shall be designed to maintain pressure integrity. Due to the different requirements and added equipment qualification for operability, it will not be feasible to change the safety classification from SC-2 to SC-1 of the portion of the main steamline from the outermost isolation valve to the turbine stop valve in Lungmen. This suggested change in classification is not required because this portion of the main steamline is not part of the RCPB, and it would not be consistent with the Turbine Building design.

Kuosheng was designed as a BWR6 plant, whereas Lungmen is a ABWR design plant. Lungmen has not downgraded the section of the main steamline (from the outermost isolation up to the turbine stop valve) and safety standard. Seismic classification of Lungmen main steamline has met the regulatory requirements, and no change is required.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-017

PSAR Section: 3.3

Question Date: March 12,1998

PSAR Question:

1. In calculating wind loads at various elevations, the report indicates they are based on Reference 3.3-1 including relevant tables for various parameter values. Please list the extent of applicability in the PSAR.
2. Since seismic Category I structures are neither slender nor flexible, wind loads can be considered as static loads loading. Please indicate whether wind loads are treated as static or dynamic loadings for stacks.
3. Wind loads for non-seismic Category I structures are based on site-specific loadings. Please explain how to determine.

Response:

1. The various tables and figures, except Table 7 and Figure 1, listed in Chapter 6 of ANSI/ASCE Standard 7-88, are applicable for the wind load analysis of buildings, components and cladding, and other structures.
2. Generally speaking, both static and dynamic wind loads are required for stack design. The dynamic loads include the effects of Karman vortex, ovalling vibration, and buffeting vibration. The static component is the normal wind load. Both components will be considered in the stack design.
3. Subsection 3.3.2.3 will be reworded as follows:

   All remainder of plant structures, systems, and components not designed for typhoon loads shall be analyzed for the site specific wind load (194.4 km/h). The methodology of calculating wind pressure will be *in accordance with Reference 3.3-1 except Table 7 and Figure 1* to ensure that their mode of failure will not effect the ability of the Seismic Category I ... safety functions.

   The PSAR will be revised as indicated in the response above. The standard number of PSAR Reference 3.3.-1 will be more accurately specified as ANSI/ASCE 7-88 (currently "-88" is missing).

ROCAEC Review Comment:

1. Taipower has responded that the wind loading will be based on ANSI/ASCE 7-88. According to "Minimum Design Loads for Buildings and Other Structures", scaled factors should be considered for wind

velocity. Please provide the scale factors for those safety and non-safety equipment. Also, what is the exposure category for SC I structures?

Further clarification:

1. The Wind velocity pressure on exposed structures are obtained from Section 6.5 of the ANSI/ASCE 7-88 reference. The equation indicated therein is based upon a 50 year return period. The ABWR SSAR Table 2.0-1 has given two basic wind speeds; a lower value, corresponding to 50 year return period, for non-safety related structures and a higher value, corresponding to 100 year return period, for safety-related structures.

   For Lungmen NPS all structures and building accessories shall be designed for 100 year return period in accordance with Bid Specifications. Thus, there is only one value of 100 year return period applicable to every structure on site. Following values are applicable to all buildings and structures at Lungmen site:

   (a) 100 year return period wind speed, V, of 54 m/s.
   (b) Importance Factor, I, equal to 1.0.
   (c) Exposure Category D
   (d) K_z, Velocity pressure exposure coefficient is obtained from ANSI/ASCE 7-88 Table 6

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

Track Number: 03-018

PSAR Sections: 3.5.1.6

Question Date: February 23, 1998

PSAR Question:

The statement that "Analysis shall be provided that will demonstrate that the probability of site proximity missiles (including aircraft) impacting the Lungmen NPS and causing consequences greater than 10CFR100 exposure guidelines is $\leq 10^{-6}$". Should 10^{-6} be 10^{-7} instead? Please clarify.

Response:

The acceptable probability of occurrence of aircraft impact is $\leq 10^{-6}$ in accordance with NUREG-0800, SRP 2.2.3 Section II, Acceptance Criteria, paragraph 4 and 10 CFR Part 100 Proposed Rule Making, page 28..

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

ROCAEC Review Comment:

Response from Taipower adopted SRP 2.2.3 which is its acceptance criteria, paragraph four, stated that the value is 10^{-6}. The third paragraph of the same section stated the value to be 10^{-7}. In SRP 3.5.1.6 on aircraft hazard, this probability is 10^{-7}. Same section in SSAR also listed it as 10^{-7}. Please clarify clearly the meanings of the various numbers.

Further Clarification (Revised 9/30/98 as agreed to in a meeting on August 21, 1998):

The calculation "Aircraft Hazard Analysis of RB, CB, and AFB; Rev. 0" (31113-0U71-1130-0001) will be re-calculated to be in full compliance with SRP 3.5.1.6, Rev. 2 (July 1981).

PSAR Section 3.5.1.6 will be revised to read as follows:

The probability of Aircraft Hazard impacting the Lungmen NPS and causing consequences greater than 10CFR100 exposure guidelines is $\leq 10^{-7}$ per year.

The fourth paragraph of PSAR Section 2.2.3.5 will be revised to read as follows:

The probability of an aircraft crashing into the plant has been evaluated, per the criteria set forth in SRP Section 3.5.1.6, Section III.2 equation, and has been shown to be acceptable and less than 10^{-7} per year.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-019

PSAR Section: 3.7.2.8

Question Date: March 12, 1998

PSAR Question:

In discussing the interactions of Seismic Category I SSCs with Seismic Category II SSCs, PSAR requires that the collapse of any Category IIC SSCs will not cause the Category IIC SSCs to strike a Seismic Category I SSCs and the collapse of any Category IIC SSCs will not impair the integrity of Seismic Category I SSCs. Since "to strike" is already not allowed, then how can it possibly "impair" the integrity ? These two requirements seem contradictory. Please clarify.

Response:

In PSAR Section 3.7.2.8, it is required that the collapse of any Category IIC SSCs will not cause it to strike a Seismic Category I SSCs <u>or</u> (Note: Not "AND") the collapse of any Category IIC SSCs will not impair the integrity of Seismic Category I SSCs.

This requirement follows that of NUREG-0800 SRP 3.7.2. For the first condition, the requirement means when a Category IIC structure topples over during a seismic event, it is prevented from <u>striking</u> a Seismic Category I structure nearby because of large physical separation between the two structures, systems or components.

On the other hand, for the second condition, the requirement means if the physical separation between the two structures is limited but the mass or size of Seismic Category IIC structure is negligible compared to that of adjacent Seismic Category I structure, collapse of Seismic Category IIC structure is not likely to <u>impair</u> the integrity of Seismic Category I structure, system or component.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-020

PSAR Sections: 3.7.1.2

Question Date: March 6, 1998

PSAR Question:

1. What method(s) will be used to develop the power spectral density functions corresponding to the site-specific design basis ground response spectra? What criteria will be used to evaluate the adequacy of the power spectral density functions?

Response:

1. The following paragraph will be added at the end of PSAR Section 3.7.1.2:

   "The site-specific power spectral density functions (PSDFs) for the horizontal and vertical components of ground motion that are consistent with the site-specific design-basis ground response spectra are generated, respectively, from the same horizontal and vertical ground-motion-time-history ensembles as those which have been used for developing the site-specific design-basis ground response spectra. The method and procedure used to generate the site-specific PSDFs consistent with site-specific design-basis ground response spectra are the method and analysis procedure recommended in the US NRC NUREG/CR-5347, "Recommendations for Resolution of Public Comments on USI A-40, Seismic Design Criteria," June 1989, including the analysis procedure recommended in Appendix B. The criteria for evaluating the adequacy of PSDFs of the design time histories will be in accordance with the criteria specified in the SRP Section 3.7.1-II.1.b Option 1 - Single Time History, supplemented with the criteria recommended in the above-mentioned NUREG/CR-5347. The comparisons of the minimum-required PSDFs with the PSDFs of the design time histories are shown in Figures 3.7-24, 3.7-25, and 3.7-26 for NS, EW, and vertical components respectively.

   These demonstrate that the PSDFs of the design time histories envelop their corresponding minimum-required target PSDFs; therefore, these design time histories satisfy the PSDF-enveloping requirements of the US NRC SRP Section 3.7.1."

   The PSAR will be revised to reflect the above response. Accordingly, Figures 3.7-24, 3.7-25, and 3.7-26 will be included and the sentence, "The Power spectral Density (PSD) check of the time histories will be provided in the FSAR," will be deleted.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-020, 03-46

PSAR Sections: 3.7.1.3

Question Date: March 6, 1998

PSAR Question:

2. NUREG-0800 requires that correlation between stress levels and damping values be demonstrated. How will this requirement be met?

Response:

2. Demonstration of correlation between stress levels and damping values is not necessary, as the RG 1.61 damping values are used, which already reflect the dependence on stress levels by having different values for the SSE and OBE. Furthermore, unless the structure is grossly oversized, the highest total stress including SSE stress in a structure is expected to be more than 50% of the material yield strength and the RG 1.61 SSE damping is adequate for this stress range. Even if the highest total stress is less than 50% of the yield strength for an oversized structure, the use of SSE damping in the SSE response analysis will not result in any safety concerns. This can be illustrated by an example. Assume the highest stress in a structure is $0.5f_y$ and it is entirely due to SSE using 7% SSE damping for concrete. Since the stress is low, the structure damping may be 4% corresponding to the OBE level. In Lungmen, the amplification factor of 7% damped horizontal ground spectra is 2.582 and the amplification factor of 4% damped spectra is about 3.34 linearly interpolated from 2% and 5% damped spectra in Figure 3.7-1. The amplification factor ratio of 4% to 7% damped spectra is thus about 1.3. The resulting stress would be $0.65f_y$ and it is still within the allowable limit. It should be noted that in this example the adjusted stress is more than 50% of the yield stress and SSE damping is again applicable. In summary, the use of RG 1.61 damping for SSE design and OBE damping for OBE design without further demonstration of correlation with stress is a technically adequate and practical design approach.

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

ROCAEC Review Comments:

2. Based on the aspect of the stress distributions of the structure, the use of the R.G. 1.61 damping values were already reflect the dependence on stress levels for SSE and OBE. For the low stresses, the lower damping

value must be adapted. The floor response spectrum will increase for the lower damping value. The results of high stresses will be built up. Thus, the correlation between stress level and damping values must be demonstrated in FSAR.

Further Clarification:

2. Correlation between stress level and damping values will be demonstrated in FSAR to show that the highest total stress in a structure is greater than 50% of ultimate strength for concrete or yield strength for steel under SSE load combinations.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **03-020**

PSAR Sections: **3.7.1.3**

Question Date: **March 6, 1998**

PSAR Question:

3. What soil material damping values will be used? How will variations of soil material damping with strain be considered?
4. Provide a technical justification for the damping values given in Table 3.7-1 for cable trays, conduits, and HVAC ductwork.
5. NUREG-0800 requires additional information on the media supporting the seismic Category I structures, including data on the soil profiles and soil properties. These data should be provided.

Response:

3. PSAR Chapter 2.5 Tables 2.5-16 and 2.5-17 show the Strain-Dependent Shear Modulus reduction and damping factors for intact and weathered rocks and for crushed rock backfill that are used to meet this requirement.
   The following will be added to Sub Section 3.7.1.3 after the last sentence:
   Chapter 2.5 Tables 2.5-16 & 2.5-17 show the strain-dependent shear modulus reduction and damping factors for the supporting medium. Response of rock to dynamic loading is discussed in Sub Section 2.5.4.7.1 for seismic Category I structures.

4. The damping values for cable trays, conduits, and HVAC ductwork are in compliance with RG 1.61 values for bolted or welded structure as appropriate.
   There is no change required in PSAR from the above response.

5. PSAR Subsection 2.5.4.2.4 lists the Engineering Properties of the foundation materials to be used on Lungmen NPS project. The existing soil profiles at the Plant site are discussed in PSAR Subsection 2.5.4.2.3.
   The following will be added to PSAR Sub Section 3.7.1.4 at the end of last paragraph:
   The existing soil profile at the plant site are discussed in Sub Section 2.5.4.2.3. Sub Section 2.5.4.2.4 lists the engineering properties of the foundation materials for use on Lungmen NPS project.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-021

PSAR Sections: 3.7.2.3.1

Question Date: March 6, 1998

PSAR Question:

1. Provide the technical justification for acceptability of using the alternative approach (i.e., the dynamic substructuring concept) to account for dynamic interaction between the subsystem and the supporting structural system. What analyses have been or will be performed to substantiate the adequacy of this approach?

PSAR Response:

1. The alternative approach to be used is based on a dynamic substructuring method to account for the effect of dynamic interaction between an equipment (equipment, component, or subsystem) and its supporting structure, i.e. the so-called 'Equipment-structure interaction (ESI)" effect. This method is based on a formulation of the equipment represented by a single-degree-of-freedom dynamic system coupled with the dynamic properties of the supporting structure at the equipment-support location represented by the equipment-support impedance (dynamic stiffness) function. The formulation and theoretical basis of this dynamic substructuring method are exact, as described in detail in Reference 3.7-8. The validity and accuracy of the method have been validated by comparing the results generated using this method with the corresponding results generated using time-history analyses of coupled equipment-structure models. This method has been applied to the Watts Bar Nuclear Power Plant and has been reviewed and accepted by the US NRC in that application in 1988.

   There is no change required to PSAR from the above response

ROCAEC Review Comments:

1. Reference 3.7-8 provides the alternative approach to account for dynamic interaction between the subsystem and the supporting structural system. This method has been applied the Watts Bar Nuclear Power Plant and has been reviewed and accepted by the USNRC in that application in 1988. Please provide the related documents of the acceptance by USNRC.

Further Clarification:

1. Please find attached copy of US NRC acceptance document for ESI application at Watts Bar Nuclear Power Plant. File NRC_ESI.PDF is attached.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-021

PSAR Sections: 3.7.2.3.1

Question Date: March 6, 1998

PSAR Question:

2. Three-dimensional structural models are preferable compared to two-dimensional models. What analyses will be performed to verify the adequacy of any two-dimensional models used?

PSAR Response:

2. To demonstrate the equivalency between a three-dimensional structural model with two-dimensional model, the natural frequencies of the two models with and without eccentricities are compared. If the frequencies are close, the two-dimensional model is considered adequate.

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

ROCAEC Review Comments:

2. The calculation of the modal participation factor includes the modal shape. The participation factors between a three-dimensional structural model and two-dimensional model are close. The modal shapes due to the different model may not be similar. Thus, the comparisons of frequency, participation factor and modal shapes are necessary to demonstrate the equivalency between a three-dimensional structural model with two-dimensional model.

Further Clarification:

2. Agree. Comparison of mode shapes will also be made.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-021

PSAR Sections: 3.7.2.3.3

Question Date: March 6, 1998

PSAR Question:

3. Specify which sections of ASCE 4-86 are pertinent to the mathematical modeling and seismic analysis of the radwaste building.
4. This section only notes that Computer Program SASSI will be used for the soil-structure interaction (SSI) analysis. Additional information to permit review of the SSI analysis with respect to NUREG-0800 requirements should be provided.
5. The discussion notes that various modifications to Computer Program SASSI have been performed. We have recently learned that use of the same nodes for the excavated soil model and interior foundation model, while implicitly permitted by the SASSI user's manual, can lead to erroneous results. In light of this, please verify that the Lungmen SASSI model is correctly developed. (See Appendix to this enclosure)
6. The PSAR notes that the seismic Category I structures are to be founded on rock. Is there a shallow soil layer over rock? Where is the control point for input of earthquake ground motion?
7. Describe how structure-soil-structure interaction between the independently founded structures has been considered.
8. NUREG-0800 does not explicitly acknowledge acceptability of the 100-40-40 method for combining maximum responses due to the three ground motion components. Provide a technical justification for using this method.
9. What are the definitions of Seismic Category IIA, IIB, and IIC structures?
10. As stated in the PSAR, constant vertical static factors were not used since all Lungmen seismic Category I structures are subjected to a vertical dynamic analysis. How is the vertical flexibility of the floor slabs of the structures accounted for in the generation of vertical floor response spectra?
11. Describe how W_p and W_b terms in Equation (3.7-11) in Section 3.7.2.14 are calculated. Also, what is the acceptable factor of safety (i.e., ratio of E_0 to E_1) for stability against overturning moments for the OBE case?

Response:

3. The USNRC Regulatory 1.143, "Design Guidance For Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants", Section 5.2 "Buildings Housing Radwaste Systems" indicates that a "simplified analysis" should be performed to determine appropriate seismic loads and floor response spectra pertinent to the location of the system.

Therefore, the mathematical modeling is anticipated to be developed following the guidance of Section 3.1 with regard to a lumped-mass stick model. Seismic analysis methods would follow the guidance of Section 3.2 for linear response-spectrum and time-history methods.

Additional data with respect to the actual model and analysis will be provided in the FSAR.

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

4. The computer program SASSI will be used for conducting soil-structure interaction (SSI) analyses for the Lungmen seismic Category I structures (Reactor Building, Control Building, and Auxiliary Fuel Building). The analyses will be conducted in accordance with the guidelines and requirements stipulated in the SRP Section 3.7.2.4 and will be based on three-dimensional SASSI models developed for the soil-structure systems of these structures. The computer program SASSI used for these analyses is the version implemented by International Civil Engineering Consultants, Inc. (ICEC) in Berkeley, California. This version of the program has been extensively validated and documented in accordance with the US nuclear quality assurance requirements.

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

5. The Lungmen SASSI models developed for soil-structure interaction analyses of the seismic Category I structures do not have the same nodes for the excavated soil model and the interior foundation model. Thus, the error condition cited in the appendix does not apply to the Lungmen SASSI analyses.

   Note: The appendix referred to in the question is not attached with the responses.

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

6. The layer of soil existing over the rock will be removed over the areas where the foundation of seismic Category I structures are to rest and the foundations will be placed on competent rock as indicated in PSAR Section 2.5.4.2.4.

   The control point for input of earthquake ground motion is the foundation level in free-field as indicated in PSAR Subsection 3.7.1.1.

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

7. The Lungmen NPS site is a rock site. All seismic Category I structures are founded on competent rock. Therefore, structure-soil-structure interaction between independently founded structures is expected to be insignificant.

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

8. The 100-40-40 method is specified in ASCE 4-86. In its commentary for Section 3.2.7.1.2 it provides technical justification as, "The 100-40-40 percent rule is based on the observation that the maximum increase in the resultant for two orthogonal forces occurs when these forces are equal. The maximum value is 1.4 times one component. As a consequence, it can be shown that the 100-40-40 rule is, in general, more conservative than the SRSS rule and is a reasonable procedure to use given the basic uncertainties involved."

   In NUREG-1462, Section 3.7.2.1, the USNRC has accepted the above approach in the final safety evaluation for a certified standard advanced PWR design of ABB-CE's System 80+.

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

9. PSAR Chapter 3, Section 3.2.1 defines Seismic Category IIA, IIB, and IIC structures. Note 'f' in Table 3.2-1d further identifies Seismic Category IIA, IIB, and IIC with appropriate design requirements for each.

10. The vertical flexibility of the floor slabs is accounted for by single-degree-of-freedom oscillators representing predominant vertical modes of slab vibration in the building seismic model.

    To make it more clear, the following sentences will be added at the end of existing sentence in PSAR Section 3.7.2.10:
    "The vertical flexibility of the floor slabs is accounted for by single-degree-of-freedom oscillators representing predominant vertical modes of slab vibration in the building seismic model. The vertical vibration properties (effective weights and vertical stiffnesses) for each mode of slab are calculated by performing the vertical eigenvalue analysis of multi-mass model of slab".

11. The energy component caused by the effects of embedment is defined as,

$$W_p = \int_0^d p_z Cz \tan\theta dz$$

where, p_z is the idealized passive soil pressure at elevation z above the base, θ is the angle of rotation at the overturning position , and C is the effective length of the structure normal to the plane of rotation and d is the depth of embedment. The work done by the buoyancy force is given by,

$$W_b = \int_{z_a}^{z_b} B_z dz$$

where za and zb are the heights of the centroid of buoyant force above the edge for equilibrium and tipping positions, respectively. B refers to the weight of the submerged portion of the structure.

The factor of safety against overturning for OBE case is 1.5.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-022

PSAR Sections:

Question Date: March 6, 1998

PSAR Question:

1. Review of Section 3.7.3 reveals only a limited number of references to OBE analyses. What types of analyses will be performed to account for differences in damping for OBE and SSE?
4. Provide a technical justification for coefficients of friction used in the analysis of frame-type supports.
5. Subsection II.9 of NUREG-0800, Section 3.7.3 requires that responses of multiply-supported equipment and components due to inertial effects and relative displacements be combined by the absolute sum, rather than SRSS. Provide a technical justification for this deviation.
6. Provide damping values for the impulsive mode of tank contents.

Response:

1. The OBE and SSE dynamic analyses are performed using the damping values contained in Table 3.7-1, which are consistent with Regulatory Guide 1.61. The dynamic analyses of OBE and SSE account for damping as described in PSAR Section 3.7.3.8.7.

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

4. The technical justification for coefficients of friction used in the analysis of frame-type supports will be provided in the FSAR. The technical justification will be based on the actual materials used and the support detail designs. The sentence in PSAR Section 3.7.3.3.4, "The following static coefficients of friction will be used ... for lubricated slide plates," will be replaced with, "The static coefficients of friction that will be used in the analysis will be provided in the FSAR with a technical justification."
   The PSAR will be revised as indicated in the response above.

5. The technical justification for combining the inertia (primary) and displacement (secondary) loads by SRSS is summarized in the last paragraph of PSAR Section 3.7.3.8.8.

   For each degree of freedom, the dynamic and the pseudo-static (displacement) contributions are time inconsistent; their peak values do not occur at the same instant in time. Therefore, it is inordinately conservative to combine the peak value of the dynamic response with the peak value of the pseudo-static response by the absolute sum method. A more representative combined peak response is provided by the SRSS combination of the peak dynamic and the peak pseudo-static values.

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

6. The damping values for the impulsive mode of the fluid-tank system will be provided in the FSAR. Item (3) of PSAR Section 3.7.3.16 will be revised as follows:

   "(3) Damping values used to determine the spectral acceleration in the impulsive mode will be based upon the system damping associated with the tank shell material as well as with the soil-structure interaction (SSI). These values will be provided in the FSAR."

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-022

PSAR Sections: 3.7.3.1

Question Date: March 6, 1998

PSAR Question:

2. Provide a technical justification for performing time-history analyses of a SRV blowdown without considering modeling uncertainties, i.e., by broadening the force time history by ±15%Dt.

Response:

2. Broadening of the input motion is used to account for uncertainties in the structural frequencies due to uncertainties in the material properties of the structure and soil (Regulatory Guide 1.122). Additionally, due to approximations in the modeling techniques used in the seismic analyses it is important to broaden the peaks associated with each of the structural frequencies. For the time-history analysis used in the piping analysis of the SRV blowdown, tests have been performed which confirm the conservatism of the analytical results. Therefore, for the SRV blowdown loads, the calculated force time-histories are not broadened by ±15%Dt.

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

ROCAEC Review Comments:

2. For the time-history analysis used in the piping analysis of the SRV blowdown, tests have been performed which confirm the conservatism of the analytical results. What are the tests?

Further Clarification:

2. Tests were performed to determine the piping responses due to safety/relief valve (SRV) opening events. These tests were sponsored by the BWR Owners' Group at a test facility in Huntsville, Alabama. A piping system was designed and constructed to determine its response to SRV opening events. This system was analyzed and tested for two cases: (1) Hydraulic snubbers used for dynamic support and (2) Rigid struts used for dynamic support. Comparisons were made of the calculated and measured support loads.

The test results and comparisons of the analysis results and measured data were presented in a paper, "Comparison of the Performances of Strut and Snubber Subject to Dynamic Load", by H.L. Hwang and E.O. Swain, Proceedings of International Nuclear Power Plant Thermal Hydraulics and Operations Topical Meeting, Taipei, Taiwan, Republic of China, October 22-24, 1984, pages J2-1 to J2-10, Published by American Nuclear Society.

The referenced paper is attached as file "SRV_TEST.PDF".

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-022

PSAR Sections: 3.7.3.3.1.4

Question Date: March 6, 1998

PSAR Question:

3. Provide a technical justification for the criteria on selection of relative anchor motions for decoupled branch pipe.

PSAR Response:

3. The decoupled branch piping has a moment of inertia of less than 1/25 of the main pipe moment of inertia, as required by PSAR Section 3.7.3.3.1.3. In addition the decoupled branch piping does not have any supports near the intersection with the main pipe that would constrain its free movement. As a result, the decoupled branch piping is relatively flexible and therefore, relative anchor motions of less than 0.16 cm can be considered to have an insignificant effect on the decoupled branch piping.

ROCAEC Review Comments:

3. The decoupled branch piping does not have any supports near the intersection with the main pipe that would constrain its free movement. Please quantify the definition of "near"?

Further Clarification:

3. PSAR Section 3.7.3.8.9(1) provides additional decoupling criteria for small branch lines. In this section the minimum acceptable distance from the run pipe to the first branch pipe support is defined.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-023

PSAR Sections:

Question Date: March 6, 1998

PSAR Question:

NUREG-0800 has not formally been changed to reflect the new OBE definition in 10CFR50 Appendix S. New plants are to be licensed in accordance with 10CFR50 Appendix S, but existing plants still are licensed to 10CFR100 Appendix A. The set of Regulatory Guides, however, clearly indicate that holders of operating licenses or construction permits issued prior to the implementation date of the active Guides (March 1997) may voluntarily implement the methods described. Thus, the USNRC has formalized the prior informal staff position regarding OBE exceedance and shutdown requirements. Provide a justification for not following the requirements in 10CFR50, Appendix S and the guidelines in Regulatory Guides 1.12, 1.166, and 1.167.

Response:

The Lungmen NPS, as summarized in PSAR Section 3.7.4, meets the guidance of Regulatory Guide 1.12, Rev. 2 , March 1997, Regulatory Guide 1.166, March 1997 and Regulatory Guide 1.167, March 1997. The necessary features to meet the guidance of these three Regulatory Guides were incorporated in the certified ABWR design when this guidance was an informal staff position on OBE exceedance and shutdown requirements. Accordingly, PSAR Section 3.7.4 meets the intent of 10CFR50 Appendix S.

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: 03-024

PSAR Sections:

Question Date: March 6, 1998

PSAR Question:

1. Section 3.8.1.4.1.1 presents a discussion on a three-dimensional finite element model of the containment structure and the reactor building. Symmetry of the containment structure and the reactor building about the centerline of the plant in a direction parallel to the fuel pool girders was considered such that only 180-degree of the structure was modeled for design analyses when subjected to the various load combinations. Explain how non-axisymmetric loads are treated when using the 180-degree model for stress analyses for wind, earthquake, and hydrodynamic loads.
2. Explain the inconsistency between Note 6 of Table 3.8-1 and Section 3.8.1.4.1.1.1 where both the SRSS method and the 100-40-40 load factor method are specified for combining co-directional effects. Since the 100-40-40 combination method is not specified in RG 1.92, provide a technical justification for its use. (Also see Question 8 on Section 3.7.2.)
3. The "S" and "U" in Note 12 of Table 3.8-1 are termed as "required strength". The ASME Section III Division 2 code uses the term "allowable stresses". Explain why the terms used in the PSAR are not consistent with the ASME code.

PSAR Response:

1. For non-axisymmetric loads the boundary displacements for nodes at the plane of symmetry follow the anti-symmetric conditions.
   There is no change required to PSAR due to above response.

2. PSAR Section 3.8.1.4.1.1.1 indicates that either of the two methods can be used on Lungmen NPS project. Note 6 of Table 3.8-1 indicates that for RCCV design, load factor (100/40/40) method is applicable. See response to Question 8 on Section 3.7.2 (Track No. 03-.21) for technical justification for the 100-40-40 combination method.
   There is no change required to PSAR due to above response.

3. ABWR SSAR Table 3.8-5 used "required strength' to define both S and U. To be consistent between Tables 3.8-1 and 3.8-5, the same notation is used in Table 3.8-1. Both terms in general, mean the maximum stress that can be allowed in the material for the given load condition.

   There is no change required to PSAR due to above response.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-025

PSAR Sections:

Question Date: March 6, 1998

PSAR Question:

1. The appendix to NUREG-0800 Section 3.8.1 states that the SRV loads should be treated as live loads in all load combinations with the exception of the combination that contains 1.5 P_a where a load factor of 1.25 should be applied to the appropriate SRV loads. The load combination No. 8 in Table 3.8-1 does not reflect this. Provide a technical justification for using a load factor of 1.0 instead of 1.25.
2. Are the concrete compressive allowable stresses given in Table 3.8-2 also applicable to primary stresses when subject to either membrane force or membrane force plus bending moment?
3. Are the v_{so} values given in Table 3.8-2 upper bound limits of tangential shear resistance that can be provided by the orthogonal reinforcement?
4. What are the values of the appropriate dynamic load factors that are used for the equivalent static loads to represent the dynamic loads resulting from a LOCA and SRV actuation (see Section 3.8.1.4.1.1.2)?
5. Describe how stress and strains in the containment wall and liner plate at the major containment penetrations will be analyzed (Subsection 3.8.1.4.1.1.3).

PSAR Response:

1. The load combination Number 8 in Table 3.8-1 follows ASME B&PV Code, Section III, Division 2, Subsection CC, Table CC-3230-1, 1989 edition, which has been reviewed and approved by the USNRC for the ABWR Standard Plant.

   The 1.25 load factor for SRV as specified in Appendix to NUREG-0800, Section 3.8.1 is a conservative value due to lack of full understanding of the SRV loading in 1970's. Since then, knowledge gained from plant tests has confirmed that the inherent margins in the SRV load definition are sufficiently conservative and further increase by using 1.25 load factor is not warranted. Thus, a more realistic factor of 1.0 as specified in the 1989 ASME Code Section III, Division 2, Sub Section CC is technically adequate.
   There is no change required to PSAR due to above response.

2. No. The allowable concrete compressive stresses are the maximum allowable corresponding to membrane force + bending moment in accordance with

Tables CC-3421-1 and CC-3431-1 of ASME B&PV Code, Section III, Division 2, Subsection CC.
There is no change required to PSAR due to above response.

3. Yes. A footnote will be added to the PSAR Table 3.8-2 indicating that the f_y and f_c' must be in MPa units.

4. These will be provided in FSAR. The second sentence from bottom of PSAR Page 3.8-9 will be revised as follows and included as the last sentence: "The attenuated pressures and equivalent static loads will be calculated based on the methodology to be presented in the FSAR."
The PSAR will be revised as indicated in the response above.

5. PSAR Subsection 3.8.1.4.1.1.3 documents the major penetrations in the containment wall. These penetrations are modeled in the global finite element model with a finer mesh. The model also includes the liner plate. The liner strains are taken directly from this global model analysis results. The analysis of stresses in the concrete surrounding these openings and the reinforcing bars therein, is performed using computer code SSDP-2D as discussed in PSAR Appendix Section 3C.3.
There is no change required to PSAR due to above response.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-026

PSAR Sections: 3.8.2

Question Date: March 6, 1998

PSAR Question:

1. In NUREG-1503, Appendix E the USNRC staff recommended using the criteria in Code Case N-284 to evaluate buckling of the ellipsoidal and torispherical drywell head under internal pressure. Provide a justification for not using such criterion in Section 3.8.2.4.1.4.
2. Define what is considered an "adequate factor of safety" for stability against compression buckling of metal components in Section 3.8.2.5.

PSAR Response:

1. Section 3.8.2.4.1.4 includes Code Case N-284 at the end of second paragraph.

There is no change required to PSAR due to above response.

2. The factor of safety complies with ASME B&PV Code, Section III, Division 1, Subsection NE, or Code Case N-284.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-027

PSAR Sections: 3.8.3

Question Date: March 6, 1998

PSAR Question:

1. Section 3.8.3.4.2 states that the concrete placed between the outer and the inner steel cylindrical shells acts to distribute loads between the shells and to provide stability to the compression elements of the reactor pedestal. Provide discussions on how the load transfer will be carried out by the concrete. Is there reinforcing steel in the concrete between the concentric steel shells? What structural elements are used to tie the concrete with the steel shells to achieve composite action?
2. Provide discussions on how the base of the reactor pedestal is structurally connected to the containment basemat through the liner plate. How are the reactor pedestal loads transferred into the containment basemat?
3. Is there reinforcing steel in the concrete between the concentric steel shells of the reactor pedestal? What structural elements are used to tie the concrete with the steel shells to achieve composite action?
4. Describe how the reactor shield wall is connected to the reactor pedestal or to the diaphragm slab.
5. Questions on Table 3.8-5:

   (1) What dynamic load factors will be used for Pa, Ta, Ra, Y, SRV, and LOCA as described in Note 3?

   (2) RG 1.92 does not include the load factor method (100/40/40) to combine the co-linear effects from multi-directional excitation components as specified in Note 7. Provide a technical justification for using this method. (Also see Question 8 on Section 3.7.2.)

   (3) In Note 12 the term "required strength" is used instead of "allowable stress" as contained in ASME Section III, Division 2. Provide a justification for deviating from the ASME Code.

   (4) Section 3.8.3.3.1 references the load combinations in Section 3.8.1.3 which includes the post-LOCA flooding. Explain why post-LOCA flooding is not one of the loads considered for the concrete internal structure.
6. For loads and load combinations considered for the steel internal structures, provide the following information:

(1) Explain why acceptance criteria for bolts of steel internal structures are not included in Table 3.8-6.

(2) Explain why post-LOCA flooding is not one of the load combinations considered.

7. Table 3.8-9 on the exceptions to ANSI/AISC N-690 is cited only in Section 3.8.4 of the PSAR. Why is it not cited in Section 3.8.3 where design of the steel internal structures is discussed?
8. Is the minimum compressive strength (27.6 MPa) of the concrete placed in the reactor pedestal and the reactor shield walls specified at 28 days or 90 days? (Subsections 3.8.3.6.2 and 3.8.3.6.3.) Are these normal weight or heavy height concrete?
9. Will bearing type or friction type A-325 high strength bolts be used for the drywell equipment and pipe support structure (DEPSS)?

PSAR Response:

1. To achieve composite action, T stiffeners are welded to the steel cylindrical shell walls on the inside faces at regular intervals. These stiffeners anchor the shells to the concrete. No reinforcing steel exists in the concrete between the concentric steel shells.

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

2. The Reactor Pedestal cylindrical wall shell plates are welded to the base plate at their lower edges. The base plate is welded to embedded stiffener plates below elevation -8200 mm. In addition, the Pedestal is anchored by anchor bolts embedded in concrete basemat.
   The transfer of load from Pedestal takes place through (1) direct bearing on base plate, (2) Shear transfer through embedded stiffener plates, and (3) by anchor bolts tension.

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

3. There are no reinforcing steel bars in the concrete within the concentric shells. For load transfer between concrete and steel shell please refer to response to Question 1 above.

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

The Reactor Shield Wall is connected to the Pedestal shell plates by full ·ration welds.
:tor shield wall vertical plates are sitting directly on top of the pedestal most vertical plates and connected to them by full penetration welds.

The diaphragm floor slab, between the elevations of 11100 and 12300, is connected to the outermost vertical plate of the pedestal and is sitting on the pedestal. Thus, the reactor shield wall is not directly connected to diaphragm floor slab.

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

5. (1) The dynamic load factors for Pa, Ta, and Ra are 1.0. For Y, the factors will depend on load distribution and the frequency of forcing functions. For hydrodynamic loads please refer to response to question 4 of Track No. 03-025
   (2) Please refer to response to question 8 of Track No. 03-021.
   (3) Please refer to response to question 3 of Track No. 03-024
   (4) Post-LOCA flooding load case is required for the design of containment structure only. It is not required for the design of containment internal structures as per the design code ACI 349 for concrete and AISC N690 for steel, nor is the SRP 3.8.3 requirement.

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

6. (1) The acceptance criteria for bolts is in accordance with ANSI/AISC N-690 as indicated in Subsection 3.8.3.2.
   (2) Please see response to Question 5 above.

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

7. In PSAR Section 3.8.3.2, applicable Codes and Standards are referenced from Table 3.8-4 which are applicable to various components of Internal Structures. For all the Internal Structures Components for which ANSI/AISC N-690 is applicable, Reference number 15 is cited. Reference 15 in Table 3.8-4 refers to modification by Table 3.8-9.

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

8. It is 90 days strength, as pozzolans are used with cement. It is normal weight concrete.

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

9. DEPSS is an all welded structure and no bolts are used in its assembly.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-028, 03-050

PSAR Sections: 3.8.4

Question Date: March 6, 1998

PSAR Question:

1. Provide in Sections 3.8.4 the descriptions, applicable codes and standards, loads and load combinations, design and analysis procedures, and structural acceptance criteria for seismic Category I aboveground and/or underground tanks.

Response:

1. The method for seismic analysis of above-ground tanks is indicated in PSAR Section 3.7.3.16. Section 3.8.4 discusses other Seismic Category I structures and covers the applicable codes and standards, loads and load combinations, design and analysis procedures, and structural acceptance criteria, which are also applicable to Seismic Category I tanks.

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

ROCAEC Review Comments:

1. We can not find the descriptions, applicable codes and standards, loads and load combinations, design and analysis procedures, and structural acceptance criteria for seismic category I aboveground and/or underground tanks in Section 3.8.4

Further Clarification:

1. Please see GE's transmittal letter GEAE-1998-0419 titled "TPC Comments on GEAE-1998-0374 (PSAR Questions 03-050 and 03-028), SC I Tanks," dated June 23, 1998.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-028

PSAR Sections: 3.8.4

Question Date: March 6, 1998

PSAR Question:

2. The last paragraph of Section 3.8.4 states that Lungmen Units 1 and 2 do not contain seismic Category I pipeline buried in soil. Are seismic Category I service water lines enclosed in an underground tunnel? If they are, provide the design considerations and acceptance criteria for both the lines and the tunnel.

PSAR Response:

2. The Seismic Category I (SC I) tanks that are located inside SC I buildings or are located in the yard will be seismically analyzed per PSAR Section 3.7.3.16 and designed in accordance with the ASME Section III Code requirements. Therefore, PSAR Section 3.8.4 does not include information on SC I tanks.

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

ROCAEC Review Comments:

2. The response discussed the piping system. What are the design considerations and acceptance criteria for underground tunnel enclosed the service water lines?

Further Clarification:

2. Section 3.8.4 states that Lungmen NPS Units 1 and 2 do not contain Seismic Category I pipelines buried in soil. Seismic Category I service water pipeline will be can run below grade from the safety related service water pump house in seismic category I pipe tunnels. Design considerations and acceptance criteria will be consistent with those applied to the safety related pump house. As indicated in PSAR section 3.8.4 and subsequent subsections, this information will be provided at the FSAR stage.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-028

PSAR Sections:

Question Date: March 6, 1998

PSAR Question:

3. Provide in Section 3.8.4 the descriptions, applicable codes and standards, loads and load combinations, design and analysis procedures, and structural acceptance criteria for the refueling pool at the top of the containment structure.
4. Is the superstructure (the portion above the fuel pool) of the reactor building a seismic Category I structure? If it is, provide in Sections 3.8.4 the descriptions, applicable codes and standards, loads and load combinations, design and analysis procedures, and structural acceptance criteria for this structure.
5. Explain why there are no discussions on the analysis and design and structural acceptance criteria for wind, typhoon, and typhoon-generated missiles in Section 3.8.4.
6. The third paragraph of Section 3.8.4.1.2 states that the Control Building is designed to provide missile and tornado protection. Is tornado a design basis load for Lungmen Units 1 and 2?
7. The first paragraph of Section 3.8.4.2.1 states that major portion of the R/B is not subjected to the abnormal and severe accident conditions associated with a containment. However, the R/B is structurally connected with the containment shell through the R/B floor slabs. Explain why the effects of the hydrodynamic loads from a SRV discharge or a LOCA in the containment are not considered in the R/B design.
8. Explain why the SRV and LOCA loads on the RB are not included in the list of loads.
9. Loads and Notations. Explain why the effects of hydrostatic pressure due to design groundwater are not discussed. Also explain how seismic-induced dynamic soil pressure is derived.
10. It appears that the "H" term in the first load combination in Section 3.8.4.3.1.2 should be "H" to represent total soil pressure (static + dynamic). Explain why the term should be "H."
11. Why is the following load combination not included in Section 3.8.4.3.1.3?
    1.5S = D + L + To + Ro (NUREG-0800 Section 3.8.4 Paragraph II.3.c.(i)(a))
12. The floor design live load of 100 lbs/ft^2 for all floors of the R/B, the C/B, and the Auxiliary Fuel Building (Section 3.8.4.3.1.1 and 3.8.4.3.2) appears to be low. Provide basis for the use of this live load for all floors.

13. Explain why the allowable stress in Subsection 3.8.4.3.5.3 for the fourth load combination under working stress design should not be "1.33S" instead of " S."
14. What are the values of the factors of safety against sliding, overturning, and flotation for the reactor building, the control building, and the auxiliary fuel building (Subsection 3.8.4.4.1)?
15. Sections 3.8.4.4.2 and 3.8.4.4.3 cited "design by rule method" as one of the methods used for design and analysis procedures for seismic Category I HVAC, cable trays, conduits and their supports. Provide a technical justification for using the "design by rule method."
16. There are no equations provided in Subsection 3.8.4.4.4.2 for checking against sliding, overturning, and flotation for the radwaste building. Provide an explanation.
17. As discussed in Subsection 3.8.4.5.2, the control building is designed per the ultimate strength method in ACI 349. Then,
    (1) Why is the term "allowable stress" used? and
    (2) Why is the stress in the rebars limited to 0.9Fy?
18. Explain why Table 3.8-10 "Exceptions to ACI 349, Appendix B Steel Embedment" is not cited in the text of Section 3.8.

PSAR Response:

3. The loads, loading combinations, and acceptance criteria applicable to Reactor Building, discussed in Section 3.8.4, are also applicable to refueling pool. Subsection 3.8.4.1.1 refers to refueling pool girders.
   The finite element model for the integrated Reactor Building and the RCCV discussed in Section 3.8.1.4.1.1 includes the refueling pool structure. The calculated forces and moments for various loads will be used for design evaluation.
   There is no change required to PSAR from the above response.
4. As indicated in Subsection 3.8.4.1.1 the superstructure of the Reactor Building consists of steel beams and columns and is a seismic Category I structure. The loads, loading combinations, and acceptance criteria applicable to Reactor Building, defined in Section 3.8.4, are also applicable to its superstructure.
   The finite element model for the integrated Reactor Building and the RCCV discussed in Section 3.8.1.4.1.1 includes the superstructure. The calculated forces and moments for various loads will be used for design evaluation of the Reactor Building superstructure.
   There is no change required to PSAR from the above response.

5. The wind load, W and typhoon load, W_t and typhoon generated missiles are defined in Subsection 3.8.4.3.1.1 and used in load combinations in Subsections 3.8.4.3.1.2 and 3.8.4.3.1.3. No separate structural acceptance

criteria is required for wind loads alone. The wind load is discussed in PSAR Section 3.3.1. The typhoon load and the typhoon generated missiles are described in PSAR Section 3.3.2. The analysis and design procedures for wind and typhoon loads are based on ANSI/ASCE 7-90 as cited in PSAR Sections 3.3.1 and 3.3.2.
There is no change required to PSAR from the above response.

6. There is no tornado design requirements for the Lungmen NPS.
   This typographical error in PSAR Section 3.8.4.1.2 will be corrected and "tornado" will be changed to "typhoon."

7. The effects of the hydrodynamic loads, from a SRV discharge or a LOCA in the containment, on the R/B structure are accounted for using an integrated R/B and RCCV analysis model.
   There is no change required to PSAR from the above response.

8. Please see response to Question 7 above.
   There is no change required to PSAR from the above response.

9. Hydrostatic pressure is treated as part of static soil pressure. Seismic-induced dynamic soil pressures are calculated from soil-structure interaction analysis.
   There is no change required to PSAR from the above response.

10. GE will correct the typographical error and will change H to H', where H' represents total soil pressure, either due to OBE or SSE event.
    The PSAR will be revised as indicated in the above response.

11. The two load cases included in Subsection 3.8.4.3.1.3 are more critical compared to the above load case since each one of them includes the above loads and include OBE in one load case and wind load in the other load case.
    There is no change required to PSAR from the above response.

12. Actual calculated Live Loads in Reactor and Control Buildings, for normal operations, are determined to be less than 100 lbs/ft^2.
    There is no change required to PSAR from the above response.

13. The allowable stress should be 1.33S.
    The typographical error in PSAR Section 3.8.4.3.5.3will be corrected.

14. The factors of safety are in accordance with NUREG-0800, SRP 3.8.5, Section II.5.

The second paragraph from bottom on PSAR Page 3.9-49 (Section 3.8.5.5) will be revised as follows:

"The allowable factors of safety of the Lungmen NPS structures for overturning, sliding and floatation are in accordance with NUREG-0800, SRP 3.8.5, Section II.5. A comparison between the allowable and the calculated factors of safety will be supplied with the FSAR.

Each foundation mat will be evaluated according to the following procedures."

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

15. Reference to "design by rule method" will be deleted from these sections. The PSAR will be revised as indicated in the response above.

16. The typographical error will be corrected. The equations are:

$$F.S._{overturning} = \frac{\text{Sum of Stabilizing Moments}}{\text{Sum of Overturning Moments}}$$

$$F.S._{sliding} = \frac{\text{Sum of Sliding Resistance}}{\text{Sum of Sliding Forces}}$$

$$F.S._{flotation} = \frac{\text{Sum of Dead Weight}}{\text{Sum of Buoyancy Forces}}$$

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

17 (1) The term 'allowable stress' is used in the general sense and indicates the maximum permissible stress allowed in the material for the specified load combination.

(2) ASME B&PV Code, Section III, Division 2, Subsection CC, Article CC-3422.1(b) allows a maximum stress in rebar of 0.9 fy. For purposes of comparing the maximum stress in rebar, this stress limit is indicated for design.

ACI 349 Code is used for the Control Building design. However, the ACI Code does not indicate how to handle concrete cracking due to thermal load. Thus, for those load cases where thermal load is significant, computer programs such as SSDP-2D (PSAR Section 3C.3), are used. The program calculates the maximum tensile stress in the rebar and compressive stress in concrete at a specific section under analysis for a given load combination. In general, the maximum rebar stress calculated by the computer program is compared with 0.9 F_y. This number is used as a guidance from ASME Code.

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

18. Section 3.8.3.2 states that Diaphragm Floor slab design shall be in accordance with Reference Number 13. Reference 13 of Table 3.8.4 lists the ACI 349 Code with the exceptions noted in Table 3.8-10.
There is no change required to PSAR from the above response.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-029, 03-051

PSAR Sections: 3.8.5.4

Question Date: March 6, 1998

PSAR Question:

1. Provide discussions in Section 3.8.5.4 on how the seismic inertia load will be transferred from the Category I structure foundation to the foundation medium. The effects of waterproofing membrane to the shear transfer should also be discussed.

PSAR Response:

1. Transfer of seismic inertia loads from the Category I structure foundation to the foundation medium is accounted for using a coupled structural and foundation model in the soil-structure interaction analysis. The effects of waterproofing membrane to the shear transfer are accounted for using a proper friction coefficient in the foundation sliding evaluation.

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

ROCAEC Review Comments:

The friction coefficient of 0.75 depends on the dewatering system. Is there dewatering system in Lungmen NPP?

Further Clarification:

Yes, a permanent dewatering system is provided at Lungmen NPS for Seismic Category I buildings in NI as indicated in PSAR Subsection 2.5.4.6.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-030

PSAR Sections: Chapter 3 Appendix 3D

Question Date: February 13, 1998

PSAR Question:

1. 3D.4.10, 3D.4.11, 3D.4.14, 3D.4.15 and 3D.4.16 only briefly mentioned the special analysis codes STEHAM, WATHAM, PILUG, PITRUST and PITRIFE without further explanation whether these codes have been verified. Please explain.
2. Section 3D.4.3 is not used. Please explain.

Response:

1. S&W analysis codes, STEHAM, WATHAM, PILUG, PITRUST and PITRIFE, are verified and documented in accordance with S&W's QA Program.

2. Section 3D.4.3 is identified as not used because the computer code that was described in the ABWR SSAR will not be used on the Lungmen project. In order to avoid potential confusion that could occur in relation to the ABWR SSAR paragraphs if all the subsequent paragraphs were renumbered, this section was maintained and identified as not used.

   Section 3D.4.3 of the ABWR SSAR described a computer code, NOZAR, that was used to perform nozzle area reinforcement calculations in accordance with ASME Code Section III NB-3640. It will not be used for the Lungmen design since NOZAR is not consistent with the ASME Code 1989. The calculations performed by NOZAR will be completed using a spreadsheet or equivalent.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-031

PSAR Sections: 3.7.1.2 Design Time History

Question Date: February 6, 1998

PSAR Question:

Please include the parameters for soil modeling. For example, number of layers, soil material properties, shear-wave velocities, dimensions of soil, etc.

Response:

The following is the information for soil profile and depth of layers for Lungmen site:

-Class A & B crushed stone Backfill
Unit 1: Depth 4.7 m (from El. 12 m to 7.3 m)
Unit 2: Depth 2.3 m (from El. 12 m to 9.7 m)

-Unit 1 weathered rock depth 2 m (from El. 7.3 m to 5.3 m)
-Unit 1 intact rock exists below El. 5.3 m.

Unit 2 intact rock exists below El. 9.7 m

The engineering properties of foundation materials, including shear wave velocity etc., are provided in PSAR Subsection 2.5.4.2.4.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-032

PSAR Sections: 3.7.1.2 Design Time History

Question Date: February 6, 1998

PSAR Question:

It is mentioned that PSD check of the time histories will be provided in the FSAR. Please include the target PSD for Lungmen in the PSAR. Is it possible to include the PSD check in PSAR review stage? If it not possible, please give the reason.

Response:

Please see response to MPR Question Number 1, Track Number MPR-3.7.1 on PSAR Section 3.7.1.2.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-033

PSAR Sections: 3.6

Question Date: February 13, 1998

PSAR Question:

OBE was not considered for the following designs :

(1) Section 3.6.1.1.1 items (1), (2) and (3)

(2) Section 3.6.2.1.4.2 on ASME code Section III, Class 2 Piping.

But after checking with SRP, OBE was found to be included for those designs above. Please clarify.

PSAR Response:

(1) OBE was inadvertently not included in items (1), (2) and (3). Section 3.6.1.1.1 will be revised such that OBE loads are included.

The following changes will be made to the PSAR:

Section 3.6.1.1.1 item (1): "(excluding earthquake loads)" will be changed to "(including OBE loads)"

Section 3.6.1.1.1 item (2): "excluding earthquake loads" will be changed to "including OBE loads"

Section 3.6.1.1.1 item (3): "including OBE loads" will be added after "equation (10)" and after "equation (13)"

(2) OBE was inadvertently not included in Section 3.6.2.1.4.2, item (d). Section 3.6.2.1.4.2 will be revised such that OBE loads are included.
The following changes will be made to the PSAR:

Section 3.6.2.1.4.2 item (d): "occasional loads" will be changed to "occasional loads including OBE"

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

ROCAEC Review Comments:

There is a footnote (+) in 3.6.1.2.4.2(1)(a) which was not explained in PSAR. Even though SSAR has explained it but it is different from B.1b(1)(a) of SRP MTP MEB 3-1 (The SRP footnote included OBE but SSAR did not include Earthquake loads). Should it be modified according to SRP ? Please explain.

Further Clarifications:

Yes, the SRP MTP MEB 3-1 footnote should be added to the PSAR. The following footnote will be added to the PSAR: "For those loads and conditions in which Level A and Level B stress limits have been specified in the design specification (including the operating basis earthquake)."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-034

PSAR Sections: 3.2 Classification of Structure

Question Date: February 10, 1998

PSAR Question:

It is mentioned (in the footnote of Table 3.2-1d) that the detailed categories of class II will be made in FSAR. Why can it be made in PSAR stage?

Response:

This question is same as Track Number 03-015, Part 1.

The detailed classification of structures, systems and components into subcategories IIA, IIB and IIC is currently in process and was not available at the time of writing the PSAR. This classification requires consideration of plant arrangement details and component functions. These subcategory classifications 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: 03-035

PSAR Sections: 3.1 Conformance with USNRC GDC

Question Date: December 2, 1997

PSAR Question:

Section 3.1.2.2.7.2

What is the applicable design requirement used for secondary containment? Please specify that in related sections. Is secondary containment a leak-tight barrier? If yes, to what extent?

Response:

The functional design requirements for secondary containment (SC) are the relevant requirements of the USNRC General Design Criteria 4, 16, and 43 and Appendix J to 10 CFR Part 50, as identified in USNRC SRP 6.2.3. The secondary containment structure together with the SC Heating, Ventilation and Air Conditioning Subsystem of the Reactor Building HVAC System (RBHV) and the Standby Gas Treatment System (SGT) provide secondary containment function which complies with these design requirements.

PSAR Section 6.2.3 provides the design bases and description of the secondary containment functional design, PSAR Section 9.4.5.1 provides a description of the SC HVAC Subsystem, and PSAR Section 6.5 provides a description of the SGT.

The secondary containment structure is not strictly a leak-tight barrier. However, an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment is established by maintaining the secondary containment at a negative pressure differential of 62 Pa (6.4 mm w.g.) when compared with adjacent regions during plant normal condition by the RBHV and during accident condition by the SGT. The plant is required to maintain secondary containment integrity and SGT operability per PSAR Chapter 16 Sections 3.6.4.1 through 3.6.4.3 and related surveillance requirements.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-036

PSAR Sections: 3.1 Conformance with USNRC GDC

Question Date: December 2, 1997

PSAR Question:

Please provide a diagram or drawing in which RCPB for Lungmen unit is clearly defined.

Response:

The definition of RCPB is provided in the second paragraph of Section 5.1 and explains how it is associated with the reactor coolant systems (RCS). The piping and instrument diagrams (P&IDs) of the RCS are listed in Section 5.1.2. The RCPB is identified in more detail in Section 5.2.4.1.1.

Section 3.1.2.2.5.2 of the PSAR will be revised to include Section 5.1, Summary Description, in the list of sections for further discussion.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-037

PSAR Sections: 3.1.2.4.2 Fracture Prevention of RCPB

Question Date: December 2, 1997

PSAR Question:

Base on the operational experiences, both our GE's BWR NPPs in Taiwan have suffered the core shroud cracks. This is also a worldwide BWRs experience. Regarding this, has any measure been taken in ABWR design to improve this potential defect therefore to have a better evaluation against GDC # 31.

Response:

Based on the current understanding of BWR core shroud cracking, there are three material factors that may act independently or jointly to produce crack initiation and propagation. These are material selection, fabrication processes, and irradiation. Older BWRs have shrouds fabricated from ordinary Type 304 material which is known to be susceptible to sensitization from the heat input of welding, and therefore susceptible to conventional IGSCC. For more recent construction, such as for Chinshan and Kuo Sheng, Type 304L has been used for core shrouds. For these cases sensitization would not be expected (as confirmed by boat sample analysis in some cases) so traditional IGSCC is not a factor. Based on evaluations and testing to date, it has been concluded that high levels of surface cold work from the fabrication of the cylindrical sections of the shroud was a likely contributor to crack initiation. In addition, for cracking observed in the beltline of some 304L shrouds, it appears that irradiation assisted stress corrosion cracking may be a contributing factor.

For the ABWR including Lungmen, all of the issues noted above have been addressed. The shroud material has been changed to a low carbon Type 316L (0.02% maximum). In addition to eliminating the potential for sensitization, this material selection brings along two other benefits. Type 316L does not form cold work induced martensite as does Type 304L, so consequently is more tolerant of surface cold working. Secondly, the 316 family of alloys appears to be more resistant to IASCC than the 304 family or, for that matter 347. With respect to fabrication practice, special measures are taken during fabrication of the ABWR shrouds to minimize residual surface cold work in the weld heat affected zones of the major structural welds. This is done by polishing the heat affected zones after welding to remove the top layer of material where most of the fabrication induced cold working is present. Also, to improve the long term resistance to IASCC design controls are included. Vertical welds in the shroud cylinder are located in low flux azimuths around the shroud circumference and the circumferential weld is located as far below the flux midplane as practical. This minimizes the end-of-life fluence of the major structural welds in the shroud. Additionally, stresses in highly irradiated weld joints are limited by a special design criteria for IASCC mitigation.

The current shroud design includes changes to address each of the three factors that are associated with current incidences of shroud cracking. Taken together, these measures result in a shroud design and fabrication that is significantly improved relative to older Type 304/304L shrouds. It is expected that these measures will be effective against the known causes of core shroud cracking.

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

Note: The criterion statement of the GDC # 31 (Title: Fracture Prevention of Reactor Coolant Pressure Boundary) addresses brittleness and adequacy of toughness of the RCPB, the low alloy steel RPV being the prime concern. It does not apply to the shroud. The stainless steel shroud belongs to the safety-related core support structures, which are addressed in PSAR Section 4.5.2. According to SRP 4.5.2, the GDC # 1 and 10 CFR 50.55a apply. The information in PSAR Section 4.5.2, as elucidated in the above response, meets the requirements of GDC # 1 and 10 CFR 50.55a.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-038

PSAR Sections: 3.3.1 Wind Loading

Question Date: November 21, 1997

PSAR Question:

1. Please provide evidence that supports the design wind velocity to be 194.4 km/h with a recurrence interval of 100 years as described in Section 3.3.1.1.

2. Same as 1 above, where is the evidence that supports the design typhoon speed to be 252 km/h as described in Section 3.3.2.1.

3. What is the unit for the wind velocity in this chapter?. Should it be fastest mile or max. of 3 minutes average?.

Response:

1. The supporting documents for design wind velocity of 194.4km/h is "CEER R81-13, The Analysis of Wind Velocity Hazard Curves at Yen-Liao Station". The report had been submitted to the ROCAEC and was accepted on December 11, 1993 via letter (82)16821 (attached).

2. PSAR Table 2.3-3, "Prevailing Wind Direction, Average and Extreme Wind Speeds in Northern Taiwan", provides the maximum instantaneous wind speed as 66.6 m/s. The actual data in this table is based on the prevailing wind directions and average / extreme wind speeds measured by the various weather stations in neighboring Taipei, Keelung, and Ilan. The references for this data are provided in PSAR Section 2.3.6.

   Kuosheng FSAR Section 3.3.2.1 gives typhoon Wind Speed of 157 mph (252km/hr). Chin Shan Nuclear Power Station FSAR design wind speed used for typhoon is 60 m/sec (216 km/hr).

   Based on above, for design of all Lungmen structures and buildings, extreme design wind speed (Reference TPC Bid Specification Section 3.3.5.1.2 - 2.1.B) of 70 m/sec (252 km/hr) is used.

3. Unit of wind velocity in this chapter is km/h. The wind velocity is the fastest speed and is not an average of 3 minutes speed.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-039

PSAR Sections: 3.5.1 Missile Selection & Description

Question Date: November 21, 1997

PSAR Question:

Section 3.5.1 indicated that the plant complies with Criterion 4 of 10CFR50 App. A. Please provide proof and further explanation.

Response:

Section 3.5.1 states, in part, "...The missile protection criteria to which the plant has been analyzed comply with Criterion 4 of 10CFR50 Appendix A, General Design Criteria for Nuclear Power Plants".

10CFR50, Appendix A, Criterion 4,"Environmental and Missile Design Basis", states, in part, "........These structures , systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles,.........."

PSAR Section 3.1.2.1.4.2, "Evaluation Against Criterion 4" has provided the following explanation:

Essential structures, systems, and components are designed to accommodate the dynamic effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, and postulated pipe failure accidents, including LOCAs.

These structures, systems, and components are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failure. The effects of missiles from sources external to the Lungmen NPS Units 1 and 2 are also considered. Design requirements specify the time which each must survive the extreme environmental conditions following LOCA. The design of these structures, systems, and components meets the requirements of Criterion 4.

The detailed discussion is provided in following sections of the PSAR:

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Chapter / Section	Title
2.0	Summary of Site Characteristic
3.3	Wind and Typhoon Loading
3.4	Water Level (Flood) Design
3.5	Missile Protection
3.6	Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping.
3.8	Design of Seismic Category I Structures
3.11	Environmental Qualification of Safety-Related Mechanical and Electrical Equipment
5.2	Integrity of Reactor Coolant Pressure Boundary
6	Engineered safety Features
7	Instrumentation and Control Systems
8	Electric Power

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-040

PSAR Sections: 3.9.2 Dynamic Testing and Analysis

Question Date: November 21, 1997

PSAR Question:

1. This section uses 60 Hz as a cut-off for RBV (Reactor Building Vibration). Would it be possible to use the pre-OP or start-up test results from KK 6/7 to verify that 60 Hz is adequate?
2. Has it been determined where to conduct the Seismic Qualification Test ? If yes, please provide the specifications items of the shaking table of the laboratory, etc.

Response:

1. The 60 Hz. cutoff frequency has no relationship to the pre-operational and startup test for K-6/7. It is the minimum cutoff frequency for hydrodynamic loads.

2. It depends on the equipment complexity as discussed in Section 3.9.2

ROCAEC Review Comments:

1. Please justify that the 60 Hz cutoff is adequate.

Further Clarifications:

1. The cut-off frequency at 60 Hz minimum is a common design practice for hydrodynamic loads in BWR Plants, and it has been used in the ABWR Standard Plant. Please also refer to PSAR Section 3.7, second paragraph on Page 3.7-1.

ROCAEC Review Comments:

The question was to ask how to verify the 60 Hz cut-off is an adequate number, considering that K6/7 is in operation (therefore there must have some test data can be referred) and the R/B size as well as embedded depth of ABWR plant are different from those of BWR4 or BWR6 plant."

Further Clarifications:

The hydrodynamic loads characteristics are the same for all GE BWR's including ABWR. The effect of building size and embedment depths is insignificant because the loading originates from the condensation loads in the suppression pool which is a common feature for all GE BWR and ABWR plants. As such, the 60 Hz minimum cutoff frequency for hydrodynamic loads based from a generic study for BWR components is applicable to ABWR.

The 60 Hz cutoff frequency has no relationship to the pre-operational and startup test for K-6/7. It is the minimum cutoff frequency for hydrodynamic loads.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-041

PSAR Sections: 3.7

Question Date: March 10, 1998

PSAR Question:

1. The May 1, 1997 edition of the ROC Building Code (with seismic resistance design regulation and commentary) specifies that the horizontal ground acceleration of the medium Seismic Zone shall be 0.23g which is higher than the OBE value of 0.2g for Lungmen. Please explain the impacts to Lungmen structure designs due to this new edition of ROC Building Codes.

2. What are the impacts to Lungmen with the new edition of the Technology Codes in the areas of soil liquefaction evaluation and prevention (chapter on Building Structures - general principles Article 48.1) and side slope stability (chapter on Building Design & Construction - buildings on slopes, Articles 260 to 268) ? Please explain.

Response:

1. The horizontal Design Ground Acceleration as specified in May 1, 1997 edition of the R.O.C. Building Code is defined as:

   $$Zd(g)=Z \times I / 1.4\alpha_y Fu$$

   Where Z = Horizontal Acceleration coefficient for individual seismic zone

   I = Important Factor

   α_y = Initial Yielding Amplification Factor

   Fu = Structural System Ductility Reduction Factor

   For a typical BOP R.C. structure located in Lungmen site, we use Z=0.23 in accordance with Chapter 2.3 of ROC Building Code; I=1.5 in accordance with Chapter 2.4 of ROC Building Code; α=1.5 in accordance with Chapter 2.6 of ROC Building Code; Fu=1.0 for conservative in accordance with Chapter 2.15 of ROC Building Code. Thus the above mentioned formula will result in:

   $$Zd(g)=0.23 \times 1.5 \ / \ 1.4 \times 1.5 \times 1.0 = 0.164g$$

   Which is smaller than 0.2g of OBE. Therefore there will be no impacts to Lungmen BOP structure designs.

2. The Seismic Category I & II Structures (Reactor Building, Control Building Auxiliary Fuel Building, and Turbine Building) are founded on rock which rules out potential liquefaction possibility. There are no slopes within the design scope of these structures.

(a) No slides have ever happened since the slopes of Lungmen site were constructed ten years before.

(b) Chapter 13 of ROC Building Regulations was promulgated on December 26, 1997. The applicable sections of Chapter 13 are 264 and 265 which concern the withdraw distance for safety reason and living quality reason respectively.

(c) Lungmen building design meets the requirements of sections 264 and 265 of Chapter 13.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-042

PSAR Sections: 3.7.1.1

Question Date: March 21, 1998

PSAR Question:

In the structural dynamic analysis, only the seismic acceleration time histories that conform to the design spectra constructed in accordance with Figures 5.1 and 5.2 of Document IESER 93002 were used. Shouldn't more seismic acceleration time histories with different phase characteristics be employed to achieve a more rational and complete analysis results ? Please explain.

Response:

Seismic acceleration time histories are developed in accordance with USNRC SRP 3.7.1, Acceptance Criteria II.1.b for Option 1, Single Time History. A single set of three-component time histories envelope both design response spectra and Power Spectral Density Function (PSDF) requirements. As compared to Option 2, Multiple Time Histories, for which a minimum of four time histories whose average response spectra envelope the design response spectra without the need to satisfy the PSDF requirements, the Option 1 of single time history is more conservative and more practical to use for design purposes.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-043

PSAR Sections: 3.2.4 Table 3.2-3

Question Date: March 19, 1998

PSAR Question:

Please explain that whether the industrial codes and standards used for the Sensing Line and Sampling Line of the Containment Monitoring System (CMS) in Table 3.2-3 under the Pipes would satisfy the requirements of R.G.1.151 ?

Response:

All the Sensing Lines and Sampling Lines are ASME Class 2 and Seismic Category I lines, and thus the industrial codes and standards used for the Sensing Line and Sampling Line of the Containment Monitoring System (CMS) in accordance with PSAR Table 3.2-3 (per column, "Pipes, Valves and Pumps") would satisfy the requirements of R.G.1.151.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-044

PSAR Sections: 3MA. 10.2

Question Date: April 15, 1998

PSAR Question:

The cooling of the RIP motor of the RCIR system is provided by the R/B cooling water system. Since RCIR is at the reactor pressure boundary, so the design of the piping of the cooling water system associated with it should follow the URS (Ultimate Rupture Strength) design in the PSAR to a value of 2.82 Mpag, but it was found that they were designed to 1.37 Mpag which is used for fire protection pipings. Please explain.

Response:

The RBCW piping is not connected to the reactor coolant pressure boundary. It is connected to the secondary side of the heat exchanger. Therefore, it is not required to be upgraded and it is designed to 1.37 MPaG.

For clarify, the last sentence of second bullet in PSAR Section 3MA.10.2 will be revised as follows:

"No upgrade is necessary for the RBCW piping since it is connected to the tube side of the RCIR heat exchangers and is not considered part of the reactor coolant pressure boundary. The design pressure for the RBCW piping is 1.37 MPaG."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-045

PSAR Sections: 3.5.1.5 and 3.5.1.6

Question Date: March 25, 1998

PSAR Question:

Site Proximity Missiles Except Aircraft and Aircraft Hazards :

NRC requested in NUREG-1503 that analysis be provided to prove that the requirements of SRP 3.5.1.5 and 3.5.1.6 are met (the consequences of missile (including aircrafts) impacts on an ABWR should not exceed the requirements of 10CFR 100 of 10^{-7} exposure per year). And GE has responded to this NRC request in SSAR (3.5.4.3 Amendment 33) but in this section of PSAR it was excluded from the Design Basis and no commitment was given that relevant analysis will be provided to prove that it does not have to be included in the Design Basis. Please explain.

Response: (Revised 9/30/98)

Please see Further Clarification of a response to Track No. 03-018 for the results of the aircraft hazards evaluation for the Lungmen NPS. The PSAR Subsections 2.2.3.5 and 3.5.1.6 will be revised as mentioned there.:

As the PSAR Subsection 3.5.1.5 mentions, events that could generate any external site-proximity missiles, other than those generated by typhoons, are considered to occur < 10^{-7} per year and, therefore, are not considered to be design basis events for the Lungmen NPS.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-046

PSAR Section: 3.7.1

Question Date: May 8, 1998

PSAR Questions:

1. Section 3.7.1.2: Provide PSAR Figures 3.7-24, 3.7-25, and 3.7-26 cited in the response to Question 1 on PSAR Section 3.7.1 (Track No. 03-020), if currently available.
2. Section 3.7.1.3: The response to Question 2 on PSAR Section 3.7.1 (Track No. 03-020) describes definition of structure damping to obtain seismic loads for structure design. However, this response does not address structure damping values for seismic response analysis to obtain floor response spectra for equipment design. How will correlation between structure damping values and structure stress levels be assured for such analyses?
3. Section 3.7.1.2: Provide numerical comparisons of the SSE ground spectra and spectra for the ground motion time-histories in the North-South, East-West, and vertical directions at 5% damping at the selected frequency intervals.

Response:

1. Figures 3.7-24, 3.7-25 and 3.7-26 are attached as FIGURES.PDF.

2. Seismic response analysis to obtain floor response spectra for equipment design is the same dynamic analysis to obtain seismic loads for structure design. Therefore, floor response spectra generated are consistent with the structure stress level.

3. Please see attached data for E-W, N-S, and Vertical response spectra at 5 % damping as TABLE.DOC File:

The PSAR will be revised as indicated in Response 1 above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-047

PSAR Sections: 3.7.2

Question Date: May 8, 1998

PSAR Question:

1. Section 3.7.2.3.1: Provide Reference 3.7-8 cited in the response to Question No. 1 on PSAR Section 3.7.2 (Track No. 03-021).
2. Section 3.7.2.3.1: The response to Question 2 on PSAR Section 3.7.2 (Track No. 03-021) indicates that only frequencies calculated by the two- and three-dimensional models are to be compared. Why are mode shapes and modal participation factors, parameters that are also important to structure seismic response, not compared as well?
3. Section 3.7.2.1: Are the main steam lines located in the turbine building? Are they required to be designed for the SSE? If so, what are the seismic design criteria for the turbine building and condenser?
4. Section 3.7.2.1: What are the criteria to be used to verify that an adequate number of masses or dynamic degrees of freedom are included in the structure models?
5. Section 3.7.2.8: Provide technical justification for Requirement No. 4 being in conformance with SRP requirements. How can it be assured that a Seismic Category IIB SSC does not fail due to the SSE, when performance under the SSE is not analyzed?
6. Section 3.7.2.12: It is stated that a comparison of responses obtained by the response spectrum and time-history analysis methods is not required since only the latter is used for Seismic Category I buildings. How is this statement consistent with Section 3.7.2.1, which states that "Analysis of Seismic Category I structures and the Reactor Pressure Vessel (RPV) is accomplished by using the response spectrum or time-history approach."

Response:

1. Reference 3.7-8, "Equipment Response Spectra Including Equipment-Structure Interaction Effects" is attached as PDF File: 3-7-2-1.PDF.

2. In addition to comparison of frequencies, comparison of modal participation factors is also made. Mode shapes are not compared separately since they are included in the modal participation factor calculations.

3.

   Main Steam Lines

The main steam lines are routed from the reactor building through the main steam tunnel (above the control building) and into the turbine building.

The MS-BOP main steam line piping, bypass line, and condenser will be used to mitigate the consequences of a postulated accident and are required to remain functional during and after an Operating Basis Earthquake(OBE) and a Safe Shutdown Earthquake (SSE). The MS-BOP main steam lines, turbine bypass valve lines including the turbine bypass valve steam chests and all branch lines 65mm(2.5in.) pipe size and larger, up to and including the first valve(including line and valve supports, safety or relief valves), that is either normally closed or capable of automatic closure during all modes of operation, shall be dynamically analyzed to demonstrate their structural integrity under OBE and SSE loading conditions.

For branch lines smaller than 65mm(2.5in) diameter, the rupture could result in bypass of the main condenser, shall be designed to withstand the SSE design loads in combination with other appropriate loads.

The turbine building is a Seismic Category IIA building (reference Table 3.2-1c). As is identified in Note f in Table 3.2-1d, the definition and basic seismic criteria for Seismic Category IIA structures are: "These are SSC whose collapse could result in loss of required function of Seismic Category I structures, components, or systems required for safe shutdown. These structures are designed according to ROC Building Code "Medium Seismicity Zone" and analytically checked to determine that they will not collapse when subjected to an SSE and extreme environmental loads. The two horizontal directions (north-south and east-west) and one vertical direction SSE accelerations are considered to act simultaneously for checking structural integrity." Also see Section 3.7.2.8.

Turbine Building and Pedestal

The Turbine Building and Turbine Pedestal is designed for earthquake loads corresponding to those specified in the R.O.C. Building Code for the medium seismicity zone. Lateral seismic loads computed based on the R.O.C. Building Code shall be applied nonconcurrently in the direction of each of the main horizontal axes of the structure. Consideration of vertical earthquake ground motion is not required.

In addition, the Turbine Building and Turbine Pedestal shall be analytically checked to determine that they will not collapse when subjected to an SSE. The two horizontal directions (i.e., N-S & E-W) and one vertical direction SSE accelerations shall be considered to act simultaneously for checking structural integrity.

Main Condenser

The main condenser is designed to withstand an SSE and maintain structural and leak-tightness integrity as required to function as the alternate leakage pathway. The design is reviewed on a realistic basis to demonstrate that it falls within the bounds of design characteristics found in conventional power plant condensers which have demonstrated good seismic performance.

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

4. The number of masses or dynamic degrees of freedom is considered adequate when it is equal to at least twice the number of significant modes in the vibration direction of interest. Alternately, the number of masses is considered adequate when additional degrees of freedom do not result in more than a 10 percent increase in responses. Significant modes of vibration are those modes up to 33 Hz or a frequency when cumulative modal mass equal to at least 90 percent of the total mass is reached.

5. As mentioned in PSAR Section 3.7.2.8, the criteria for Seismic Category IIB SSC is in accordance with USNRC Regulatory Guide 1.143, Revision 1, "Design Guidance For Radioactive Management Systems, Structures, and Components Installed In Light-Water-Cooled Nuclear Power Plants." Seismic design for radwaste management systems and structures housing radwaste management systems is addressed in Section C.5 of this Regulatory Guide. The maximum earthquake required to be considered is the OBE; not the SSE.

   Seismic Category IIB SSC are located a sufficient distance away from Seismic Category I SSC that the collapse or failure of the Seismic Category IIB SSC will not result in the loss of required function of Seismic Category I SSC required for safe shutdown.

6. The intent of PSAR Section 3.7.2.1 is to document that both the response spectrum and the time history methods are acceptable for dynamic analysis of Seismic Category I (SC I) structures, which include not only buildings but also tanks, tunnels, dams, etc. The purpose of PSAR Section 3.7.2.12 is to document that all SC I *buildings* use time history methods.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-048

PSAR Sections: 3.7.3

Question Date: May 8, 1998

PSAR Question:

1. Section 3.7.3: This section does not reveal any information on computer codes used for subsystem analysis nor benchmark problems to verify codes. Please describe the computer codes and the verification processes to be used.
2. Section 3.7.3: This section does not reveal any displacement limits for piping. Please describe displacement criteria to be used to assure the performance of hangers and other types of supports.
3. Section 3.7.3.8.8: It is stated that differential endpoint or restraint deflections result in secondary stresses in the piping systems because the stresses are self limiting much as the case for restraint of thermal expansion. Because of the secondary nature of the pipe stresses and the fact that the primary stresses due to inertial response of the piping system and the secondary stresses due to differential endpoint or restraint displacements are dynamic in nature and are not expected to occur at the same time, the two responses are combined by SRSS. This argument is based upon the dynamic response and structural behavior of piping but does not necessarily apply to the reactions at piping supports nor to support design philosophy. The loads that are considered self limiting for piping design are usually treated the same as loads that are not self limiting for support design. Please describe how the forces/stresses in piping supports due to seismic inertia and support point displacement are combined for support design and the justification for the combination.

Response:

1. The computer codes used for subsystem analysis include the following codes:

   NUPIPE-SWPC - Used to perform piping system analysis.
   STRUDL-SW - Used to perform structural frame analysis.
   PC-PREPS - Used to perform structural frame, weld, local stress, and baseplate and anchor bolt analysis.
   ANSYS - Used to perform analysis of many types of subsystems

   Computer codes used for subsystem analysis are described in Section 3D. Verification of these computer codes is described in Section 3D.1. For example, PISYS is the computer code used for analyzing piping systems subjected to both static and dynamic loads. The PISYS computer code and the benchmark problems used to verify it are described in Section 3D.4.1.

2. Section 3.9.4.1 provides the displacement criteria to be used to assure the performance of hangers and other types of supports.

   There are no specific displacement limits for piping. Displacement limits are governed by other plant and system commodities. For example, piping displacements must be limited such that unacceptable thermal or seismic interactions between piping and other plant commodities are precluded. Displacements must be limited so as to not allow unacceptable pipe sag or the formation of water pockets that may be result in unanticipated water or steam fluid transient loads or events.

   The performance of hangers and other types of supports will be guaranteed by the proper selection of support and hanger type, location and function in accordance with pertinent Code and vendor requirements after consideration of all applicable loadings and pipe movements.

3. The pipe support loads due to seismic inertia and support point displacements are combined by SRSS.

   For each support direction, the dynamic and the pseudo-static (displacement) contributions are time inconsistent; their peak values do not occur at the same instant in time. Therefore, it is inordinately conservative to combine the peak value of the dynamic response with the peak value of the pseudo-static response by the absolute sum method. A more representative combined peak response is provided by the SRSS combination of the peak dynamic and peak pseudo-static values.

   The forces/stresses for pipe supports due to seismic inertia and support point displacement are combined in accordance with criteria as stated in PSAR Sections 3.9.3.4 "Component Supports," Section 3.9.3.4.1 "Piping," Table 3.9.1 "Plant Events," and Table 3.9-2 "Load Combinations and Acceptance Criteria for Safety Related ASME Code Class 1, 2 and 3 Components, Component Supports and Class CS Structures." These sections and tables state that for pipe supports the loading combinations for various operating conditions correspond to those used for design of the supported pipe. The tables state that the seismic anchor motions (SAM) are dynamic loading events and that they are to be combined in accordance with NUREG-0484, Revision 1. This NUREG permits, as is stated in PSAR Section 3.7.3.8.8 "Effect of Differential Building Movements," the combination of dynamic loads by the Square Root Sum of Squares (SRSS) method. The justification is provided in those documents and it is concluded that the peak load values are not expected to occur at the same time and that the computation of the SAM effects from static analysis uses a method in which the displacements are applied to produce the most conservative loads. Other supporting justification is that the dynamic response time for earthquake functions is rapidly varying, the duration of the strong motion portion of the earthquake function is short, the earthquake function consists of a few distinct high peaks which are random with respect to time, and the earthquake response is calculated on a linear elastic basis.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-049

PSAR Sections: 3.8.1

Question Date: May 8, 1998

PSAR Question:

1. In the response to Question No. 5 on Section 3.8.1 (Track No. 03-025), TPC stated that these penetrations are modeled in the global finite element model with a finer mesh. The model also includes the liner plate. The liner strains are taken directly from this global model analysis results. The analysis of stresses in the concrete surrounding these openings and the reinforcing bars therein, is performed using computer code SSDP-2D as discussed in PSAR Appendix Section 3C.3. Further questions are as follows:
   a. Explain how major penetrations such as the upper drywell equipment and personnel hatches, the lower drywell equipment and personnel tunnels and hatches, and the suppression chamber access hatch, which are oriented at different azimuths will be modeled in the 180-degree global model (Section 3.8.1.4.1.1).
   b. Explain how concrete cracking (Section 3.8.1.4.1.1.3) will be accounted for in the linear elastic analysis of the global finite element model of the containment structure.

Response:

1. a. The RB global FE model includes the southern half (azimuth 0°-90°-180° region) of the RCCV. All major penetrations on the RCCV wall are modeled at the actual locations except for the upper drywell personnel hatch, which is located on the northern half of the RCCV. The azimuth angle of the upper drywell personnel hatch is at 230 degrees, which is nearly symmetrical with the upper drywell equipment hatch at 130 degrees azimuth. This means the U/D personnel hatch is indirectly modeled in the global model. Therefore, it can be concluded that all major penetrations are considered in the 180 degree global model.
   b. Concrete cracking is accounted in the calculations by SSDP-2D. SSDP-2D calculates the stresses of concrete and rebars under the condition that concrete does not bear tensile forces. Section forces and moments are obtained by the RB global FE model analyses. The global model analyses are elastic finite element analyses, and do not consider concrete cracking.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-050

PSAR Sections: 3.8.4

Question Date: May 8, 1998

PSAR Question:

1. TPC response to Question No. 1 (Track No. 03-028) referred to Section 3.7.3.16 for seismic analysis and Section 3.8.4 for applicable codes and standards and structural acceptance criteria. Review comments on TPC response are:
   a. Provide descriptions of the Lungmen Seismic Category I tanks (aboveground and underground) and their designs and constructions. This is consistent with what has been presented for the other Lungmen Seismic Category I civil structures.
   b. The ASME Code Section III, Subsection NE, Division 1, Class MC listed in Section 3.8.4.2 is not applicable for aboveground atmospheric steel storage tanks.

   Seismic analysis method discussed in Section 3.7.3.16 is only applicable to Seismic Category I above-ground vertical flat bottom storage tanks. Provide discussions for Lungmen underground Seismic Category I tanks, if applicable.

2. Provide calculations of the "actual" floor live loads for the floor with the heaviest live load of the Reactor Building and the Control Building to substantiate the use of 100 lbs/ft2 live load.

Response:

1. a. See response to MPR Question 03-028, Part 1, on PSAR Section 3.8.4. Free-standing Seismic Category I (SC I) tanks are divided into two categories: (1) Those tanks that are located inside SC I buildings are seismically analyzed per PSAR Section 3.7.3.16 and designed to the ASME III requirements, and (2) those tanks that are located in the yard will be seismically analyzed per PSAR Section 3.7.3.16 and designed to the requirements to be provided in the FSAR, Section 3.8.4.

   b. As mentioned in the above response the free-standing tanks that are located inside Seismic Category I (SC I) buildings are designed to the ASME III, Subsection ND, requirements; Subsection NE requirements do not apply.

There are no Seismic Category I aboveground and/ or underground tanks in the BOP portion of the plant. Therefore Sections 3.8.4 is not required.

2. The live load during normal plant operation, in general, consists of people walking or any light machinery on the floor. The magnitude of this load is used as 100 lbs/ft^2. This is based upon judgment aided by ASCE Standard ASCE 7-88, "Minimum Design Loads for Buildings and Other Structures" Table 2. The live load during plant refueling is equal to the higher of 100 lbs/ft^2 or the expected floor loading during refueling.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-051

PSAR Sections: 3.8.5

Question Date: May 5, 1998

PSAR Question:

1. We agree with TPC's response to Question No. 1 on Section 3.8.5 (Track No. 03-029) that the coupled structural model of the building and the foundation will properly model transfer of the load between the building and the foundation. However, describe how the loads will be physically transferred from the building into the foundation medium. What are the proper coefficients of friction between the concrete mat and the membrane and between the membrane and the bedrock for design?

Response:

1. For the buildings the foundation construction is a multi-layered approach. On top of the intact rock a layer of 100 mm thick mudmat will be placed. This layer will have drainage piping embedded and will have its surface intentionally roughened. A water proof membrane will be laid on top of this layer of mudmat. On top of membrane layer will be another 100 mm thick layer of mudmat. The basemat will be placed on top of this mudmat layer.

   The coefficients of friction between various materials are as follows:

Coefficient of Friction	Static	Dynamic
Between mudmat & bedrock/Rockfill	0.5	0.75
Between concrete & non-vulcanized butyl Rubber sheet	0.75	>0.75

   Non-vulcanized butyl rubber sheet will be used as waterproofing membrane. Based upon various text books the coefficient of friction of rubber to concrete ranges from 0.6 to 0.9 assuming a smooth surface. Actual value will be higher because of the presence of irregular surface between mudmat and rubber membrane. This is consistent with Kuo Sheng FSAR values.
   A friction coefficient of 0.75 takes into account the degradation due to the presence of a waterproofing membrane considering that the surfaces above and below the membrane are not smooth.
   Based upon this the governing coefficient of friction will be between mudmat and bedrock/rockfill.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-052

PSAR Sections: 3.2.2

Question Date: May 4, 1998

PSAR Question:

1. The Safety Classifications of Reactor Internal Structures and Core Support Structures defined in the Reactor Pressure Vessel System in 3.2-1(b) number B11 are not consistent with the classifications in Table 3.2-3 or Core Support Structures in Section 3.9.5.1.1 or Internal Structures in Section 3.9.5.1.2. Please clarify.

2. Comparing the systems listed in Table 3.2-1a with SSAR, PCV Pressure Leak Test Facility and Motor Control Center, etc., are missing. Please clarify.

3. In SSAR Table 3.2-1, item number J1 "Fuel Assembly", Loose Part Monitoring System was classified but it was not mentioned in PSAR Table 3.2-1. Please clarify.

4. In Table 3.2-1 item number K11, the relevant piping at the Containment Boundary of the Radwaste Sump was classified as SC-2. Should the Sump itself be classified as SC-2 as well ? Please clarify.

5. The Standpipe of the Condensate Storage & Transfer System listed in Table 3.2-1 item number P13 was classified as SC-2 component but the connecting piping was not safety-related component. Please clarify.

6. The Ultimate Heat Sink listed in Table 3.2-1C item number W11 did not have descriptions of mechanical components and pipings associated with it. Please clarify.

7. The pipings within Quality Group B under Quality Group & Seismic Category of Figure 3.2-2 did not have its specifications listed after its Branch Line is connected to the downstream valve. Please clarify.

PSAR Responses:

1. The PSAR Table 3.2-1b will be revised to be consistent with Table 3.2-3 and PSAR Sections 3.9.5.1.1 and 3.9.5.1.2 as attachment 1.

2. SSAR System T25 (Sequence No. T5 on SSAR Page 3.2-49; see Page 3.2-15 for sequence no. and system no. correlation), PCV Pressure and

Leak Testing Facility, is not included in the Lungmen NPS design at this time, because a consideration is being given if this special facility area in secondary containment would be needed. The FSAR would include the system if it is decided that the facility would be required.

SSAR System R24 (Sequence No. R7), Motor Control Center is included in PSAR System R12, Low voltage Distribution System on PSAR Page 3.2-40. The PSAR System R12 is comprised of SSAR Systems R23 (Power System) and R24 (Motor Control Center) and includes additionally non safety-related items.

The preceding explanation is an example of system restructuring for Lungmen PSAR from the SSAR system structure. A Lungmen system (1) may not exist similar to an SSAR system because it is not needed, or (2) it may be a new system that does not exist in SSAR, or (3) it may be same as the SSAR system, or (4) it may be a combination of portions of some SSAR Systems.

The tables in attachment 2 show the relationship matrix of those PSAR systems which are same as or combination of portions of some SSAR Systems. If a PSAR system shown on PSAR Table 3.2-1b is not in the matrix column "Lungmen PSAR Table 3.2-1b System," it is a new system; and if an SSAR system on SSAR Table 3.2-1 is not in the matrix column "SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System," it is not included in the PSAR.

3. The Loose Parts Monitoring System (LPMS) is System C75 in the Lungmen PSAR.

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

4. The Radwaste Sumps (SUMP) System (System designation: K11) is classified as a non-safety-related system. Its function is to collect radioactive liquid wastes generated in the plant and transfer the waste water to the Radwaste Building for further processing. However, part of the SUMP piping penetrates through the primary containment boundary. This containment penetration, including piping and associated isolation valves, forms an integral part of the containment boundary and are therefore classified as the same safety class as the primary containment boundary, i.e., SC-2. The remainder of SUMP has no safety design basis and is classified as non-safety -related.

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

5. The standpipe is classified as SC-2 since it supports the safety-related function of the condensate storage tank level transmitters. The safety function of these transmitters is to ensure a source of reactor coolant inventory water (suppression pool) is available by switching the suction for the emergency core cooling systems from the condensate storage tank to the suppression pool if the condensate storage tank is unavailable. Failure of the connecting pipe to the standpipe will not prevent the transmitters from performing their safety-related function, since the transmitters will detect the unavailability of the condensate storage tank and switch the suction to the suppression pool.

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

6. The Ultimate Heat Sink (UHS) is neither a building nor a system , it is the ocean , and has no mechanical components or piping associated with it. Therefore , the UHS (W11) will be deleted from Table 3.2-1c.

   The heat removal capability for safety related systems is provided by the Reactor Building Service Water System -P26 (RBSW) which withdraws from then discharges water to the ocean. This system is listed on pages 3.2-27 & 3.2-59 of Tables 3.2-1b & 3.2-1c with the description of the mechanical components and piping. It should be noted that the pumps and filters associated with the RBSW are in the BOP scope and located in the Reactor Building Service Water Pump House of the Intake area of the Lungmen NPS.

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

7. The Quality Group and Seismic Category of the portion of branch piping is Quality Group C and Seismic Category I. The branch piping is a simplified representation of the feedwater system interface with the RWCU system. Figure 3.2-2 will be updated to show the change from Quality Group B to Quality Group C.

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

REVISED Attachment 1

Table 3.2-1b Classification Summary - Nuclear Island

Principal Component[a]	Safety class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
B11 Reactor Pressure Vessel System (RPV)						
ITEMS 5, 6, 7 and 8 ARE REVISED AS FOLLOWS:						
5. Reactor internal structures:						(jj)
· Safety-related internals - spargers: feedwater, RHR shutdown cooling low pressure flooder, and high pressure core flooder systems	2	C	—	S	I	
· Safety-related internals - Core support structures	2	C	—	S	I	
· Safety-related internals (except spargers and core support structures)	3	C	—	S	I	
· Non-safety related (ther") internals	N	C	—	R/G	II	
6. Reactor Internal Pump Motor Casing (a part of RPV boundary)	1	C	A	S	I	

Note (jj) to be added to Table 3.2-1d (see last column above, Item 5):

jj. The design requirements of reactor internal structures (except core support structures) are not available in the ASME Section III Code. Therefore, they are not identified on Table 3.2-3. Refer to Subsection 3.9.5.3.6 for the requirements.

Note to be added to Table 3.2-3: § See Note (jj) in Table 3.2-1d for design requirements of the reactor internals, other than the core support structures presented in this table.

Lungmen PSAR Table 3.2-1b System	SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System
B11	B11
B11	F13
B11	F14
B21	B21
B21	N11
B21	N39
B31	B31
B31	F14
B31	F17
B31	F22
C11	C11
C12	C12
C12	C94
C12	F17
C12	F21
C31	C31
C41	C41
C51	C51
C61	C61

Lungmen PSAR Table 3.2-1b System	SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System
C71	C71
C73	E31
C75	J11(3)
C81	C81
C82	C82
C85	C85
C91	C91
E11	E11
E22	E22
E51	E51
F11	F11
F12	F32
F13	F12
F14	F16
F15	C93
F15	F15
F22	U47
F23	U48

Lungmen PSAR Table 3.2-1b System	SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System
F31	U31
F32	U32
F41	F51
F43	R51
G31	G31
G41	G41
G51	G51
G61	N25
G61	N26
G61	N27
G62	P91
G63	P91
H11	H11
H12	H12
H14	H14
H21	H21
H21	H22
H21	H25

Lungmen PSAR Table 3.2-1b System	SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System
H23	H23
J11	J11
J12	J12
K11	K17
K11	U46
K12	K17
K12	P91
K13	K17
K68	N21
K68	N62
N11	N36
N12	N36
N13	N22
N14	N35
N15	N37
N21	N21
N21	P92
N22	B21

Lungmen PSAR Table 3.2-1b System	SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System
N22	N21
N22	N38
N31	N31
N32	N32
N33	N33
N34	N34
N41	N41
N41	N42
N43	N43
N44	N44
N51	N51
N61	N61
N62	N72
P11	P10
P11	P11
P13	P10
P13	P13
P16	U43

Lungmen PSAR Table 3.2-1b System	SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System
P18	U42
P21	P21
P22	P22
P24	P24
P25	P25
P26	P41
P27	P42
P28	N71
P51	P51
P51	P81
P52	P52
P54	P54
P61	P62
P62	P61
Not Included	P63
P71	P32
P72	P95
Not Included	P73

Lungmen PSAR Table 3.2-1b System	SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System
Not Included	P74
R10	R10
R10	R11
R10	R13
R10	S12
R11	R21
R11	R22
R12	R23
R12	R24
R13	R46
R14	R47
R15	R52
R16	R42
R21	R43
R21	Y52
R31	R40
R41	R34
R51	R31

Lungmen PSAR Table 3.2-1b System	SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System
T22	T22
Not Included	T25
T31	T31
?	T41
T41	U41
T42	U41
T43	U41
T44	U41
T45	U41
T49	T49
T61	D21
T61	D23
T62	B21
T62	D23
T62	P91
T62	T10
T62	T53
T63	D11

Lungmen PSAR Table 3.2-1b System	SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System
U71	R35
U71	T10
U71	T11
U71	T12
U71	T13
U71	U71
U71	U21
U71	Y31
U72	U24
U72	U72
U73	U73
U73	U21
U74	U74
U74	Y31
U75	U75
W11	P40
Y86	Y86

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Lungmen PSAR Table 3.2-1C System	SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System
B21	B2
B21	N13
C31	C3
C85	C10
F21	N/A
F23	U9
F24	N/A
F31	U3
F41	F12
G61	N/A
G62	P20
H14	H2
H21	H3
H21	H4
K11	K1
K12	N/A
K13	N/A
K14	N/A
K68	N22
N11	N12

Lungmen PSAR Table 3.2-1C System	SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System
N12	N12
N13	N3
N14	N11
N21	N2
N21	N4
N21	N5
N21	N6
N22	N2
N23	N3
N31	N7
N32	N8
N33	N9
N34	N10
N35	N/A
N41	N16
N42	N/A
N43	N18
N44	N19
N51	N20
N61	N21

Lungmen PSAR Table 3.2-1C System	SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System
N62	N24
P11	P0
P11	P1
P13	P2
P16	U6
P17	N/A
P18	U5.1
P19	N/A
P20	N/A
P22	P4
P24	N/A
P26	P9
P27	P10
P28	N/A
P29	N/A
P30	N/A
P31	N/A
P51	P11
P52	P12
P54	P13

Lungmen PSAR Table 3.2-1C System	SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System
P56	P19
P61	N/A
P62	N/A
P71	P7
P72	P22
R41	P9
R51	P8
S21	N/A
S31	R16
T30	N/A
T32	N/A
T33	N/A
T34	N/A
T35	N/A
T36	N/A
T37	N/A
T38	N/A
T39	N/A
T41	N/A
T42	N/A

Lungmen PSAR Table 3.2-1C System	SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System
T44	N/A
T45	N/A
T47	N/A
T48	N/A
T50	N/A
T55	N/A
T56	N/A
T57	N/A
T58	N/A
T59	N/A
T61	N/A
T63	N/A
T64	N/A
T66	N/A
T68	N/A
T69	N/A
T70	N/A
T71	N/A
T72	N/A
U72	U11

Lungmen PSAR Table 3.2-1C System	SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System
U74	U13
U81	N/A
U82	N/A
U84	N/A
U85	N/A
U87	N/A
U91	N/A
U92	N/A
U93	N/A
U94	N/A
W11	P8
W12	N/A
W13	N/A
Y11	N/A
Y12	N/A
Y21	N/A
Y41	N/A
Y42	N/A
Y51	N/A
Y53	N/A

Lungmen PSAR Table 3.2-1C System	SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System
Y54	N/A
Y86	Y3
Y99	Y1

Lungmen PSAR Table 3.2-1C System	SSAR Table 3.2-1 System(s) that are contained totally or partially in PSAR System

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-053

PSAR Sections: 6.2.3.4.1

Question Date: May 27, 1998

PSAR Question:

Figure 6.2-38 was said to be supplied until FSAR but it is a figure that describes general principles and would be more appropriate to be supplied during PSAR. Please supply this figure.

Response:

Figure 6.2-38 will be included in the revised PSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-054

PSAR Sections: Ch. 3

Question Date: Apr 29, 1998

PSAR Question:

It is requested that TPC commit a time table to provide the relevent design/analysis reports of Lungmen pressure vessel, core components and Reactor Internal pumps and FMCRD at the pressure boundary to show that 10CFR50 regulations and relevent industrial codes are met. TPC is requested to commit a time table to provide design and analysis reports for the following components :

1. Reactor Internal pump under RPV Penetration analysis
2. FMCRD penetration analysis
3. HPCF Sparger
4. Core plate
5, Core shroud
6. Feedwater inlet/outlet nozzle
7. Main steam nozzle
8. Head Spray and vent nozzle
9. Biological shield related components

Response:

1. Reactor Internal pump under RPV Penetration analysis
   Design Spec.(see Note 1) is a part of the RPV design spec which is already issued and will be forwarded to the ROCAEC.
   Stress Report (see Note 1) date: will be available by the time of equipment delivery (see Note 2)

2. FMCRD penetration analysis
   Design Spec. is a part of RPV design spec (same as the spec in Item 1) which is already issued and will be forwarded to the ROCAEC.
   . Stress Report date: will be available IN FORM OF VPF (see Note 3) by the time of equipment delivery

3. HPCF Sparger
   Design Spec. date: 8/98 (see Note 2)
   Stress Report date: will be available by the time of equipment delivery

4. Core plate
   Design Spec. date: 8/98

Stress Report date: will be available in form of VPF by the time of equipment delivery

5. Core shroud
   Design Spec. date: 8/98
   Stress Report date: will be available in form of VPF by the time of equipment delivery

6. Feedwater inlet nozzle
   Design Spec. is a part of RPV design spec which is already issued and will be forwarded to the ROCAEC.
   Stress Report date: will be available in form of VPF by the time of equipment delivery

7. Main steam nozzle
   Design Spec. is a part of RPV design spec which is already issued and will be forwarded to the ROCAEC.
   Stress Report date: will be available in form of VPF by the time of equipment delivery
8. Head Spray and vent nozzle
   Design Spec. date: 10/98
   Stress Report date: will be available by the time of equipment delivery
9. Biological shielding of the RPV is provided by the Reactor Shield Wall which is part of the containment internal structures as described in PSAR Section 3.8.3. A summary of the structural design analysis will be provided in FSAR.

Note (1): In the design specification, the design requirements of the equipment/component are defined. In the stress report, the results of the stress analyses (mechanical and thermal) of the equipment/component are provided.

Note (2): TPC will transmit the design specifications as they will be available by the dates shown. The stress reports will be transmitted as they will be available by the time of equipment delivery.

Note (3): VPF (Vendor Print File - a term used for a document to be prepared by a component vendor)

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-055

PSAR Sections: App. 3L

Question Date: May 4, 1998

PSAR Question:

1. The design of pipe whip restraint in SSAR was based on REDEP and its database but in PSAR 3L.3.1, no mention was made of such database. Please explain if other methods were used.

2. The simulated analysis model described in 3L.3.2 quoted standard models in Figures 3L-1 and 3L-2 but the figures were not found in the report. Please provide those figures.

Response:

1. The original design of GE's U-Bar pipe whip restraint is based on the computer program REDEP. The results of this program were used to develop a pipe whip restraint property table, not an electronic database, which is applicable to GE's U-Bar pipe whip restraint design. This program was not mentioned in the PSAR since REDEP will not be used for the Lungmen Project.

2. Figures 3L-1 and 3L-2 were inadvertently left out of the PSAR. The Figures 3L-1 and 3L-2 will be added to Appendix 3L.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-056

PSAR Sections: 3.6.2.1.4.3

Question Date: May 18, 1998

PSAR Question:

1. The postulated locations of breaks of Class 1 piping in this section are not consistent with the rules in SRP 3.6.2 MEB3-1B.1c(1). Please clarify.

2. During piping re-analysis, the conditions for determining intermediate break locations are not consistent with SRP 3.6.2 MEB3-1B.c(i), (ii), (iii). Please clarify.

Response:

1. Section 3.6.2.1.4.3 is correct and consistent with SRP 3.6.2 and MEB 3-1, as modified by the exception to MEB 3-1B.1.c(1)(b) stated in Section 3.6.1.1.1(3).

2. During piping re-analysis, the conditions for determining intermediate break locations are consistent with SRP 3.6.2 revision 2, MEB3-1B.c(i) and (ii).

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-057

PSAR Sections: 3.6.2.1.4.2

Question Date: May 18, 1998

PSAR Question:

1. The stress value listed in (1)(d) was 0.8(1.8Sh + SA) which is different from SRP's 0.8(1.2Sh + SA). Please clarify the difference.

2. The stress value was the lesser of 2.25Sh and 1.8Sy in (1)(e) which is different from 1.9Sh in SRP. Please clarify.

Response:

1. The stress value listed in (1)(d) is in accordance with SRP 3.6.2 revision 2, MEB3-1.

2. The stress value listed in (1)(e) is in accordance with SRP 3.6.2 revision 2, MEB3-1.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-058

PSAR Sections: 3.6.2.1.6.1

Question Date: May 18, 1998

PSAR Question:

1. Does the break described in (1) mean circumferential break only or it includes axial break as well ? Please clarify.

2. In the postulated pipe breaks, circumferential breaks can be neglected for pipings under 25mm and axial breaks can be neglected for pipings under 100mm. But from the requirements in SRP, those should still be considered when the maximum stress range exceeds the stipulations in 3.6.2.1.4.3 and 3.6.2.1.4.4. This section is unclear on this part and does not conform to the requirements of SRP. Please clarify.

Response:

1. The break described in (1) is only for circumferential breaks. In item (1), "No breaks" will be replaced by "No circumferential breaks".

2. Per Section B.3.a(1) of SRP 3.6.2 revision 2, MEB3-1, circumferential breaks can be neglected for piping having a nominal diameter less than or equal to 25 mm.

   Per Section B.3.b(1) of SRP 3.6.2 revision 2, MEB3-1, longitudinal breaks can be neglected for piping having a nominal diameter less than 100 mm.

   In addition, to clarify that item (9) of Section 3.6.2.1.6.1 applies only to longitudinal breaks, the following change should be made to the first sentence. Replace the words "The dynamic" with "For a longitudinal break, the dynamic".

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-059

PSAR Sections: 3.6.2.1.6.2

Question Date: May 18, 1998

PSAR Question:

This section starts out by describing the criteria that are used to postulate cracks in high- or moderate-energy fluid systems but the criteria listed were only for moderate-energy pipings and no mention was given to high-energy pipings. Is it missing ? Please clarify.

Response:

Section 3.6.2.1.6.2 provides the criteria used to postulate cracks in high- and moderate-energy fluid systems. The following changes will be made to clarify this:

In the beginning of Section 3.6.2.1.6.2, the words "high- or moderate -energy" will be replaced with "high- and moderate-energy".

In item (1) of Section 3.6.2.1.6.2, the words "moderate-energy" will be deleted since this item applies to both high- and moderate-energy fluid system piping.

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

Track Number: 03-060 (renamed from 03-061)

PSAR Sections: 3.9.5

Question Date: May 4, 1998

PSAR Question:

1. The descriptions of the loading and load combination of RPV internals are scattered in different sections of PSAR and attention should be given to its consistency. Please clarify if the following statements are consistent :

   (1) 3.9.1.4.3 rules that internals will only check the faulted load of service level D. Is it the same as in the plant event 7, 8 and 9 in Table 3.9-2 ? Please clarify.

   (2) But according to Table 3.9-2, internals will have to undergo fatigue analysis which is contrary to the requirement that only faulted condition should be evaluated in 3.9.1.4.3. Please clarify.

   (3) Table 3.9-7 required that the usage factor of internals should be ≤ 1.0 in service level A and B. Does this fatigue analysis include the 50 times (cycles) listed for plant events 1, 2, 3, 4, 5, 6, 7 and 12 of Table 3.9-1 ? Please explain.

2. The spargers of the safety system are also part of the internals whose fluid load is obviously different from the other internal structures. The major accidents considered for them are SBL, IBL and LBL only. Is this enough ? Please clarify.

3. From the shroud safety evaluations experience gained at Chinshan and Kuosheng, for the Section XI evaluation, it was found that under upset condition, load was seriously impacted. Is it enough to consider (only) faulted condition evaluation in 3.9.1.4.3 ? Please clarify.

Response:

1.

   (1) All Core Support Structures are evaluated for all loading conditions and ASME service levels (A, B, C, & D) as shown in PSAR Table 3.9-2, including fatigue analyses. The PSAR Section. 3.9.1.4.3 discusses <u>in particular</u>, per the format required by Regulatory Guide 1.70, Rev. 3, the bases for the faulted condition for core support structures and other safety related reactor internal components. It does not mean that core support structures are evaluated for faulted condition (ASME service level D) only.

   (2) Fatigue usage evaluation is required for all core internal components that are classified to be ASME-III, Class CS structures. Please note that the title of Table 3.9-2 includes class CS structures type of internals. These reactor internal

components are listed in Section. 3.9.5.1.(1). Section 3.9.1.4.3 is not the ONLY required analyses for those equipment as explained above.

(3) Please note that Table 3.9-7 is applicable only to safety class reactor internal structures only. So, the plant events listed in Table 3.9-1 are considered in the usage factor determination as they are applicable to an individual internal component. According to Footnote 2 of Table 3.9-1, Plant Events 1, 4, 5 and 9 apply only to the RPV. In accordance with Footnote 7, which applies only to Plant Event 12, OBE, fifty OBE cycles are used in the evaluation. The safety class reactor internal structures are evaluated accordingly.

2. The fluid loads are included in the stress analyses for the three sparger systems (feedwater, HPCF and LPCF). These loads include the flow induced vibration (FIV) loads and the jet reaction loads. The analysis results show no significant stresses are induced by these loads. There are no other significant fluid loads on the spargers. The fluid loads are identified in the definition of "Normal (N)" load in PSAR Table 3.9-2. The jet reaction (JR) load will be included in the definition.

3. All Lungmen CS structures are analyzed in accordance with the loading conditions and acceptance criteria of Table 3.9-2, not just the faulted condition. Please see responses to Question 1 above, which clarifies this.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-061

PSAR Sections: 3.8.4.2.7, 3.8.4.3.6, 3.8.4.4.5, 3.8.4.5.7

Question Date: 6.30, 1998

PSAR Question::

For the safety service water pump house, Lungmen PSAR provided a paragraph of description of the structure in Subsection 3.8.4.1.7. Other PSAR information concerning applicable codes, standards and specifications, loads and load combinations, design and analysis procedures, and structural acceptance criteria for the pump house are to be provided in the FSAR as stated in PSAR Subsections 3.8.4.2.7, 3.8.4.3.6, 3.8.4.4.5 and 3.8.4.5.7, respectively. It is likely that the design considerations of the safety service water pump house will be similar to that of other seismic category I structure. As a minimum, design commitments should be stated in the PSAR.

PSAR Response:

The PSAR (Sections 3.8.4.2.7, 3.8.4.3.6, 3.8.4.4.5, 3.8.4.5.7, and 3.8.4.6.5) will be revised as follows:

3.8.4.2.7 Safety Service Water Pump House

The safety service water pump house shall be designed using the following codes and standards:

(1) ACI 349, Code Requirements for Nuclear Safety-Related Concrete Structures.

(2) ANSI/AISC-N690, Specification for the Design, Fabrication and Erection of Steel Safety-Related Structures for Nuclear Facilities.

(3) AWS, Structural Welding Code, AWS D1.1.

(4) NRC Regulatory Guides:

   (a) Regulatory Guide 1.28 – Quality Assurance Program Requirements (Design and Construction)

   (b) Regulatory Guide 1.29 – Seismic Design Classification

   (c) Regulatory Guide 1.61 – Damping Values for Seismic Design of Nuclear Power Plants

   (d) Regulatory Guide 1.94 - Quality Assurance Requirements for Installation Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants

   (e) Regulatory Guide 1.142 – Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containment)

(5) ANSI/ASCE 7 – Minimum Design Loads for Buildings and Other Structures

(6) Steel Structures Painting Council Standards

   (a) SSPC-PA-1 Shop, Field Maintenance Painting

(b) SSPC-PA-2 Measurement of Paint Film Thickness with Magnetic Gages
(c) SSPC-SP-1 Solvent Cleaning
(d) SSPC-SP-5 White Metal Blast Cleaning
(e) SSPC-SP-6 Commercial Blast Cleaning
(f) SSPC-SP-10 Near-White Blast Cleaning

(7) Applicable ASTM Specifications for Materials and Standards

3.8.4.3.6 Safety Service Water Pump House

Refer to the loads, notations and combinations established in Subsection 3.8.4.3.1.1~3.8.4.3.1.3, except that accident pressure Pa, and pipe break loads Yr, Yj, Ym do not exist.

Stress free design temperature is 22°C for portion above sea water level.
Stress free design temperature is 18°C for portion under sea water level.

Air temperature inside building are as follows:

Summer operation	40°C
Winter operation	10°C

3.8.4.4.5 Safety Service Water Pump House

The Safety Service Water Pump House will be designed in accordance with ACI 349 for concrete structures and ANSI/AISC-N690 specification for steel structures.

The Safety Service Water Pump House will be analyzed using the computer programs: ABAQUS, SAP90 and etc.

The analysis and design results will be supplied with the FSAR, these will include safety factors against sliding, overturning, and flotation.

3.8.4.5.7 Safety Service Water Pump House

Structural acceptance criteria are defined in the ANSI/AISC-N690 for steel structures and ACI 349 Codes for concrete structures. In no case does the allowable stress exceed 0.9Fy, where Fy is the minimum specified yield stress for steel. The clearances between adjacent buildings are sufficient to prevent impact during a seismic event.

The typhoon load analysis methodology for this building is the same as that for the Reactor Building. The results of the analysis will be supplied with the FSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-062

PSAR Section: 3.8.4

PSAR Question:

A subsection which provides information on the materials used in the construction of the reactor building, control building, auxiliary fuel building, and radwaste building needs to be provided in the PSAR. The subsection should include information such as major materials of construction (e.g. reinforced concrete, structural steel and anchors) quality control, special construction techniques, and quality assurance.

PSAR Response:

The concrete materials for construction of RCCV are indicated in Subsection 3.8.1.6 and the steel components of RCCV in Subsection 3.8.2.6. For Containment Internal Structures the construction materials are specified in Subsection 3.8.3.6.

Similarly, a new Subsection 3.8.4.6 will be added to the PSAR as follows:

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

3.8.4.6.1 Reactor Building

The concrete materials used in Reactor Building shall be in accordance with ACI 349 and structural steel shall be as defined in ANSI/AISC-N690.

The specified compressive strength of concrete at 91 days shall be at least 34.5 MPa, except for basemat, which shall be at least 27.6 MPa.

Reinforcing steel shall meet ASTM A615, Grade 60.

ASTM A572 / A36 structural steel shall be used for construction.

Concrete anchors shall be welded Nelson studs.

Structural bolts, studs, and nuts with diameter > 19 mm shall be ASTM A325.

Structural bolts, studs, and nuts with diameter < 19 mm shall be ASTM A307.

3.8.4.6.2 Control Building

The materials, quality control, and construction techniques used for the control building are the same as those used for the reactor building in Subsection 3.8.4.6.1.

3.8.4.6.3 Auxiliary Fuel Building

The materials, quality control, and construction techniques for auxiliary fuel building are the same as those used for reactor building in Subsection 3.8.4.6.1.

3.8.4.6.4 **Radwaste Building**

The concrete material used in Radwaste Building shall be in accordance with ACI 318-95 and structural steel shall be as defined in AISC 9th edition.

The specified compressive strength of concrete at 28 days shall be at least 27.6 Mpa.

Reinforcing steel shall meet ASTM A615, Grade 40& Grade 60.

ASTM A36 structural steel shall be used for construction.

Concrete anchors shall be welded Nelson studs.

Structural bolts used in structural steel connections shall be ASTM A325 high-strength carbon steel bolts for bearing type connection, except where ASTM A490 bolts are specifically noted. Minimum size shall be ¾ inches (19mm)

Anchor bolts shall be ASTM A307.

Welded connection material conforms to the requirements of AWS D1.0 for E70XX electrodes.

3.8.4.6.5 **Safety Service Water Pump House**

The concrete materials used in Safety Service Water Pump House shall be in accordance with ACI 349 and structural steel shall be defined in ANSI/AISC-N690.

The specified compressive strength (fc') of concrete at 28 days shall be at least 27.6 Mpa.

Reinforcing steel shall meet ASTM A615, Grade 60.

ASTM A572/A36 structural steel shall be used for construction.

Concrete anchors shall be Arc Welding Stud.

High strength bolts shall be ASTM A325.

Anchor bolts shall be ASTM A307.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-063

PSAR Section: 3.8.4

PSAR Question:

The turbine building is a seismic Category IIA structure. Per definition in Table 3.2-1d, seismic Category IIA are SSC whose collapse could result in loss of required function of seismic Category I structures, components, or systems required for safe shutdown. This the turbine building must be capable of withstanding the Lungmen Safe Shutdown Earthquake (SSE) so as not to impair safety function of a portion of the main steam line, the condenser, or the adjacent seismic Category I structures. There are currently no discussion in Section 3.8.4 concerning design commitments for the turbine building. Subsections on codes and standards, loads and load combinations, structural acceptance criteria, design and analysis procedures of the turbine building should be provided.

Response:

The basic criteria for the Turbine Building, as it relates to safety related structures, is that it not collapse as a result of the safe shutdown Earthquake (SSE). This basic criteria is provided in Note of Table 3.2-1d for seismic category IIA structures. i.e., "These structures are designed according to ROC Building Code. "Medium Seismicity Zone" and analytically checked to determine that they will not collapse when subject to an SSE and extreme loads.".

Therefore, there is no change required to the PSAR from the above responses.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-04-001

PSAR Sections: 4.3.2.1 Shutdown Reactivity

Question Date: October 15, 1997

PSAR Question:

Reference 4.3-3 gives the uncertainty of cold core calculations, but it was presented in 1977. GE12 is a new advanced fuel not covered in the reference. Please describe the uncertainty of cold core calculations for GE12 and provide enough data to support it.

PSAR Response:

Two reloads of GE12 are currently operating. The following is a summary of the cold critical eigenvalues obtained with these reloads.

Plant (BWR Type)	Cycle	Projected Cold critical k_{eff}	Actual Cold critical k_{eff}
Fitzpatrick (BWR/4 D lattice)	13	.9997	1.0028
Perry (GBWR/6 C lattice)	7	.9970	.9959

The actual cold critical values obtained are in good agreement with the projected cold critical values. The projected values are based on current methods experience gained with 8x8 and 9x9 fuel for each plant. Therefore the GE12 cold criticality evaluations should have the same uncertainty as the previous experience. This total experience indicates a standard deviation in the cold critical eigenvalue of 0.33%. See also the GE response to question Track Number I-04-003.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-04-002

PSAR Sections: 4.3.2.1 Shutdown Reactivity

Question Date: October 15, 1997

PSAR Question:

In this subsection, it is described that the core must be capable of being made subcritical, with margin, in the most reactive condition throughout the operating cycle with the highest worth control rod "or" any control rod pair with same Hydraulic Control Unit (HCU), fully withdrawn and all other rods fully inserted. It seems to be possible that only one control rod with the highest worth fully withdrawn was taken in some calculations. Under what conditions you took the highest worth control rod? If there are no calculations with the highest worth control rod, please erase this piece.

PSAR Response:

The ABWR design has 205 control rods with 103 HCUs. Each two control rods share one HCU except for the central rod which has its own HCU. It is possible that the central rod can be the highest worth control rod for certain core design. Therefore, the words, "the highest worth control rod or highest worth control rod pair," should be used. See also the GE response to question Track Number 04-005.

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

Track Number: I-04-003

PSAR Sections: 4.3.2.1 Shutdown Reactivity

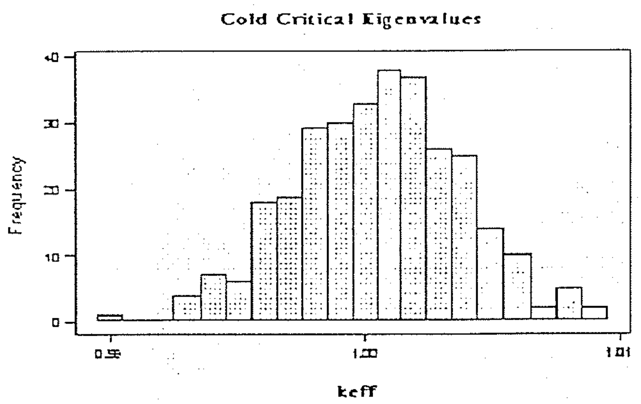
Question Date: October 15, 1997

PSAR Question:

The long cycle with the maximum 18 months is required in the Lungmen Units, but the uncertainty of cold core calculations given in Reference 4.3-3 did not cover it. Please describe the uncertainty of cold core calculations for long cycles and provide enough data to support it.

PSAR Response:

The following data base represents cold critical predictions accumulated over the past seven years and represents cycle length ranging from 12 months to 24 months. Bundle average enrichments range from 3.2% to 4.2%, which bound the enrichments projected for the Lungmen core for 18 month cycles. This data base consists of 306 cold critical tests conducted at 38 plants. The average calculated critical eigenvalue for this database is 1.0005. The standard deviation for the data is 0.33% (.0033) and represents the uncertainty in the cold critical reactivity for the Lungmen core design. See also the GE response to question Track Number I-04-001.



There are no changes required to the PSAR based on the response above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-004

PSAR Sections: 4.1 Reactor - Summary Description (4.1.4.1 Reactor Internal Components)

Question Date: December 2, 1997

PSAR Question:

In GE SSAR, there are nine computer codes used for reactor internal component analyses. However, there are just three of them being used for Lungmen PSAR. What are the rest of the six codes for?

PSAR Response:

The SSAR used the following computer codes for reactor internals component analyses;

1. NASTR04V
2. SAP4G07
3. HEATER
4. USAGE01
5. ANSYS
6. CLAPS
7. ASIST
8. SEISM03
9. SASSI

The PSAR in turn used only the following computer codes for reactor internals analyses;

1. SAP4G07
2. ANSYS
3. SEISM03

It was thought that the other six codes would not be required for the Lungmen NPS design. However, as discussed below, four of the codes will not be retained in the PSAR but two will be:

a) The computer codes USAGE01, CLAPS and ASIST were used for the ABWR reactor internal component analyses. A comparison of these codes to ANSYS has shown that the analytical capabilities of these codes can all be covered by ANSYS which has better post-processing features. Therefore, these codes are replaced by ANSYS for the Lungmen reactor internal component analyses.

b) SASSI is applied to perform dynamic soil-structure interaction analysis whose results (output from the code) are used as input to the reactor internal component analysis code such as ANSYS or SAP4G07. Presently the SASSI code is described under the Soil Structure Interaction in Section 3C.5 and a duplicate description is not needed in Section 4.1.4.1.

c) Since Heater was used for the initial hydraulic design input for the ABWR feedwater sparger, a new Section 4.1.4.1.4 HEATER, will be added to reinstate the HEATER code as follows:

4.1.4.1 Reactor Internal Components

Computer codes used for the analysis of the internal components are as follows:
(1)SAP4G07
(2)ANSYS
(3)SEISM03
(4)HEATER

4.1.4.1.4 HEATER

HEATER is a computer program used in the hydraulic design of feedwater spargers and their associated delivery header and piping. The program utilizes test data obtained by GE using full-scale mockups of feedwater spargers combined with a series of models which represent the complex mixing processes obtained in the upper plenum, downcomer, and lower plenum. Mass and energy balances throughout the Nuclear Steam Supply System (NSSS) are modeled in detail. The program is used (1) in the hydraulic design of the feedwater spargers for each BWR plant, (2) in the evaluation of design modifications, and (3) the evaluation of unusual operational conditions.

d) Upon reconsideration, GE thinks that NASTR04V may continue to be used for the Lungmen FMCRD analysis. Therefore, the description of NASTR04V will be included more suitably in Section 3D.2, FMCRD, under Subsection 3D.2.2, as follows:

3D.2.2 Structural Analysis Programs

Structural analysis programs, such as NASTRO4V (described in this section) and ANSYS (described in Subsection 4.1.4.1.2), are used in the analysis of the FMCRD.
NASTRO4V is a GE in-house version of the MSC/NASTRAN program (Digital VAX Version 64) which is developed and maintained by the MacNeal Schwendler Corporation in Los Angeles. As a general purpose computer program for finite element analysis, its capabilities include (1) static response to concentrated and distributed loads, to thermal expansion and to enforced deformation; (2) dynamic response to transient loads, to steady-state sinusoidal loads and to random excitation; (3) determination of real and complex eigen values for use in vibration analysis, dynamic stability analysis, and elastic stability analysis; (4) nonlinear static and dynamic analysis including material and geometric non-linearities; and (5) steady-state and transient heat conduction.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-005

PSAR Sections: 4.3 Nuclear Design (4.3.2.4.1 Shutdown Reactivity)

Question Date: December 2, 1997

PSAR Question:

In the calculation of the shutdown reactivity in Section 4.3.2.4.1, the terms "strongest control rod" and "strongest control rod pair" are used in Table 4.3-2 and the equation, respectively. The terms are also mixed in this section. Please make some clarifications.

PSAR Response:

The design basis for selection of control rod pairs with the same HCU, is that the control rods are neutronically decoupled. This results in a negligible change in core reactivity with one or both control rods fully withdrawn, assuming all other control rods are fully inserted. Therefore, the analysis performed for the PSAR only assumed a single, high worth control rod was fully withdrawn. Withdrawal of the paired control rod would not impact the calculational results.

The PSAR Section 4.3.2.4.1 will be modified as follows:

The following will be added after the first sentence of the first paragraph:

"The design basis for selection of control rod pairs with the same HCU, is that the control rods are neutronically decoupled. Therefore, the difference in core reactivity with one or both control rods fully withdrawn, assuming all other control rods are fully inserted is negligible."

Further Clarification to ROC-AEC's comments:

The design basis for shutdown margin (SDM) calculation is to consider the worst single failure, which is a failure of one hydraulic control unit (HCU). The ABWR design has 205 control rods with 103 HCUs. Each two control rods share one HCU except for the central rod which has its own HCU. Therefore, a failure in one HCU causes a rod pair or a rod (the central rod) to fail. In order to cover the worst case, the SDM calculation should consider the highest worth control rod (the central rod) or highest worth control rod pair (with same HCU) fully withdrawn and all other control rods fully inserted.

Based on this , the PSAR will be revised as follows:

4.3.1.1 Reactivity Basis

The nuclear design shall meet the following basis: The core shall be capable of being made subcritical at any time or at any core condition with the highest worth control rod or highest worth control rod pair (with same HCU) fully withdrawn and all other control rods fully inserted.

4.3.2.4.1 Shutdown Reactivity

Revise second sentence to read, "The shutdown margin is determined by using the BWR simulator code (see Section 4.3.3) to calculate the core multiplication at selected exposure points with the strongest rod or strongest rod pair with same HCU fully withdrawn"

Revise the equation to read,

k_{eff} = k_{eff} (Strongest rod or Strongest rod pair with same HCU withdrawn) $_{BOC}$ + R

Revise last paragraph to read, "The calculated values of k_{eff} with the strongest rod pair withdrawn at BOC and of R are reported in Table 4.3-2. For completeness, the uncontrolled k_{eff} and fully controlled k_{eff} values are also reported in Table 4.3-2. (It should be noted that the central rod, which has its own HCU does not have the strongest worth in this design.)

Table 4.3-2 Calculated Core Effective Multiplication and Control System Worth - No Voids, 20°C

Change "Strongest Control Rod Out" to "Strongest Control Rod Pair Out"

Change asterisk * footnote to read "For the core loading in Figure 4.3-1, Critical k_{eff} = 0.995."

Figure 4.3-1 Equilibrium Core Loading Map

Change core coordinates shown from J to I and I to J.

16.1.1 Definition for Shutdown Margin (SDM)

Revise Item (c) as follows:

(c) All control rods are fully inserted except for the control rod or control rod pair of highest reactivity worth, which is assumed to be fully withdrawn.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-006

PSAR Sections: 4.4 Thermal-Hydraulic Design

Question Date: December 3, 1997

PSAR Question:

1. In Figures 4.4-1 and 4.4-2, there are maximum allowable core flow lines that constitute the boundaries of the flow map, yet it lacks text description in Section 4.4.3.3.1. Please add it on.

2. Please clarify the maximum rod line of flow map for the Lungmen plant. Is it the same for ABWR SSAR of 102%?

PSAR Response:

1. The text in Section 4.4.3.3.1 will be modified to reflect that the core flow limits are 111% (10 of 10 RIPS) and 103% (9 of 10 RIPs).

   The second sentence of Section 4.4.3.3.1 will be modified as follows: "The power-flow map for 10 RIP operation (maximum 111% flow) is shown in Figure 4.4-1, and for 9 RIP operation (maximum 103% flow) in Figure 4.4-2."

2. The maximum rod line for Lungmen NPS with GE12 core design was defined as that rod line passing through the 100% Power at 85% core flow condition. Thus the ABWR SSAR 102% rod line is not the same as the maximum rod line for the Lungmen PSAR.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-007

PSAR Sections: **4.3.2.2.2 Power Distribution Accuracy**

Question Date: December 2, 1997

PSAR Question:

The methodology used in the power distribution accuracy in Lungmen design is different from that used in the SSAR. Please explain. (Notice the different reference sources: Reference 4.3-2 "Xenon Consideration in Design of Boiling Water Reactor, APED-5640, 1968", as opposed to the SSAR "Steady-State Nuclear Methods, NEDO-30130-AII, April 1985.")

PSAR Response:

In the SSAR, Reference 4.3-2, Steady-State Nuclear Methods, NEDO-30130-A, April 1985 was referenced in Section 4.3.2.6.1, Xenon Transients. This reference was not proper for xenon transients and therefore, a correct reference was included in the PSAR as Reference 4.3-2, Xenon Consideration in Design of Boiling Water Reactor, APED-5640, 1968. However, since the PSAR Section 4.3.2.2.2 still depends upon the original reference, the following changes will be made:

Section 4.3.2.6.1 Xenon Transients

Last sentence, change to read, "Analysis and experiments conducted in this area are reported in References 4.3-4 and 4.3-7."

Section 4.3.5 References

Change Reference 4.3-2 back to the original SSAR reference and add new Reference 4.3-7 as follows:

4.3-2 Steady-State Nuclear Methods, NEDO-30130-A, May 1985.
4.3-7 R. L. Crowther, Xenon Consideration in Design of Boiling Water Reactor, APED-5640, 1968.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-008

PSAR Sections: **4.2.2.2 Control Rods**

Question Date: December 2, 1997

PSAR Question:

It is mentioned in the text that either boron carbide or hafnium will be used as control rod material in the Lungmen plant. What are the major merits and defects of the two materials?

PSAR Response:

The Lungmen NPS initial core will feature two control rod types, the Duralife 230 and the Duralife 120. The Duralife 230 is a long-life design which features B_4C in high purity, tantalum-stabilized stainless steel absorber tubes, Hafnium absorber in the three outer edge rod locations, and Hafnium absorber at the tip of the control rod. The Duralife 230 control rods will occupy the control cell positions used for power shaping. The Duralife 120 control rod is a long residence time control rod that features high purity, tantalum-stabilized stainless steel absorber tubes and all B_4C absorber. The Duralife 120 control rod will occupy the shutdown positions in the reactor and is not intended for power shaping operation.

B_4C is the preferred absorber material because of its high neutron cross section relative to hafnium, its lower weight and its relatively lower cost. Upon irradiation, however, B_4C swells according to the reaction $^{10}B + {}^{1}n \rightarrow {}^{7}Li + {}^{4}He$. Thus for absorber locations that incorporate B_4C enclosed in a tube as a poison, the tube lifetime may be limited by B_4C swelling, resulting in tube cracking. Hafnium does not swell upon irradiation. Hafnium sheet / strip / rod is therefore substituted for B_4C in those high burnup regions of power shaping control rod designs in order to extend control rod life.

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

Track Number: 04-009

PSAR Sections: **4.3.2.2.3 Power Distribution Anomalies**

Question Date: **December 2, 1997**

PSAR Question:

It is mentioned in the text that a specific margin in the peaking factor to account for power tilt is unnecessary. Does that same conclusion hold when the APR is in the auto mode?

PSAR Response:

The Automatic Power Regulator (APR) functions to automatically maneuver the plant through normal criticality's, heatup, power operation, and cooldown operations. The APR automatically controls either generator or reactor power by changing control rod positions and/or core flow.

No additional penalty is needed on the core thermal limits nor does the reactor operator need to be especially cautious when operating on the APR.

The reason is the existence of the ATLM (automatic thermal limit monitor) which is "normalized" by 3 D monicore (the three dimensional power distribution program normally used by the operator to monitor thermal limits). This program on request or automatically downloads thermal limit information to the dual ATLMs which then automatically monitor the core through power maneuvers while operating on the APR. Should thermal limits be exceeded (by a conservative approximation algorithm), either ATLM will trip the plant into APR manual and halt any rod or flow movement.

Since the same thermal limits are involved that are applicable to manual operation and the APR is restricted to operating on predefined control rod sequences stored in the RC&IS and since the APR is constrained to operate on predefined power/flow map trajectories, there will be no more of a power tilt operating automatically than manually.

Additionally it should be known that for automatic operation, all rod movements use symmetric ganged rod movements that will promote having symmetric core power distributions within the reactor core and not the possibility of significantly asymmetric power distributions. The same is true for semi-automatic rod movements. However, in the manual mode of rod movement, the operator (above the Low Power Setpoint) has the capability to set asymmetric rod patterns, if so desired, but generally speaking this is not a desired operating strategy. The ABWR core design is not prone to flux tilt problems due to the large damping effect of the strong void reactivity influence.

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

Track Number: 04-010

PSAR Sections: 4.3.2.4.1 Shutdown Reactivity

Question Date: December 2, 1997

PSAR Question:

It is very confusing to have "strongest rod" and "strongest rod pair" intermingled both in the text and the formula. Please clarify.

PSAR Response:

Please see response 04-005.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-011

PSAR Sections: 4.4.3.3.3 Regions of the Power/Flow Map

Question Date: December 2, 1997

PSAR Question:

Based on our understanding of the Chapter, the region III in the Fig. 4.4-1 should read "operation within this region is precluded by SCRRI". This, however, is not consistent with the footnote in Fig. 4.4-4, please explain.

PSAR Response:

Figure 4.4-4 Stability Controls and Protection Logic provides the two following logic functions: (1) With reactor power ≥30% power, core flow ≤ 36% flow, and two or more RIPs tripped, SCRRI is initiated to reduce reactor power by control rod insertion to below the least damped region (Region III). (2)With reactor power ≥25% power, core flow ≤ 36% flow, and two or more RIPs tripped, a rod block signal is generated to preclude re-entry into Region III by control rod withdrawal.

Figures 4.4-1 and 4.4-2 will be redrawn to show Region III bounded by the natural circulation line, maximum rod line "A", the RIP minimum flow line "1", and the 30% power line. Region I will be extended up to the 30% power line. The PSAR Figures 4.4-1 and 4.4-2 will be revised accordingly.

The notes on Figure 4.4-4 will be revised as follows:

1. SCRRI is bypassed below 30% power (analytical limit), as 30% power corresponds to the approximate power level on the 60% rod line at natural circulation conditions.

2. SCRRI is bypassed above 36% core flow (analytical limit), as 36% core flow corresponds to the approximate core flow with 8 RIPs operating at minimum speed above 30% power.

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

Track Number: 04-012

PSAR Sections: **4.4.3.3 Power/Flow Operating Map**

Question Date: December 2, 1997

PSAR Question:

The text says that the maximum rod line comprises the operating point at 100% power and 85% flow, which is obviously not the case in Fig. 4.4-1. Please explain.

PSAR Response:

The maximum rod line as shown on Figure 4.4-1 should pass through the 100% power and 85% flow point (start of the horizontal portion of the maximum rod line "A"). The power to flow map was generated directly from the RODAN VAX code whose hard copy output was converted into a figure for the PSAR. The scaling of the figure was inadvertently shifted by the word processing personnel.

The PSAR Figure 4.4-1 will be redrawn with the correct scaling.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-013

PSAR Sections: **4.4.3.7 Thermal Hydraulic Stability Performance**

Question Date: December 2, 1997

PSAR Question:

Regarding the stability issue, please provide the following information:

1. A comparison of channel stability performance between Lungmen NPP and standard ABWR.
2. A comparison of core stability performance between Lungmen NPP and standard ABWR.

PSAR Response:

Lungmen NPP will be using the GE12 fuel design while the standard ABWR when first deployed used the P8x8R fuel design. The GE12 fuel is designed to improve fuel economics, while maintaining stability margin as the experienced based P8x8R fuel. It is shown in the GE12 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II), NEDE-32417P, December 1994, report that channel decay ratio for the GE12 fuel is lower than that of the P8x8R fuel under identical operating conditions, therefore, GE12 fuel provides additional margin to the onset of regional oscillations. The core decay ratios are marginally greater for the GE12 fuel, but are well within the uncertainty of stability methods. As demonstrated in the GE12 Compliance with Amendment 22 report, this results in a negligible impact on the stability performance of the plant.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-014

PSAR Sections: 4.5, Reactor Materials

Question Date: December 2, 1997

PSAR Question:

1. Are there any specification limits regarding the hardness, bending or stress/strain of the austenitic stainless steel? Please explain.

2. In the CRD system structural material Section 4.5.1.1 (a), Spool Piece Assembly, please provide information regarding the material specification of the thrust bearing. In addition, please explain whether all CRD system components meet all related codes and standards.

4. Please explain the difference between the two units (Ferrite Number and percentage) used in explaining the delta ferrite contents (PSAR Section 4.5.2.4). Is it possible to use just one unit? Is there difference between these two units then, the delta ferrite limit of 8%-20% specified in the PSAR for the austenitic stainless steel does not agree with the 8FN-20FN limit specified in the GE SSAR. Please clarify.

PSAR Response:

1. In general, there are no code or specification limits on cold work in austenitic stainless steel materials. However, some of the ASTM/ASME material specifications (A/SA-240 Plate, A/SA-249 Tube) do have material hardness limits. The GE material specifications require a maximum limit on material hardness. The materials are used in the solution annealed condition. The fabrication specifications limit the amount of induced cold work by specifying maximum hardness, minimum bend radii, and maximum induced strain during fabrication.

   In addition, in response to TPC Bid Specification, Appendix A Chapter 1, paragraph 5.3.1.1, GE requires the following for cold work control of wrought austenitic stainless steels for service above 200°F(93℃),or which are part of the reactor coolant pressure boundary:

   - Hardness of austenitic stainless steel raw materials will not exceed 92 HRB.

2. The proposed material of construction for the thrust bearing is SUJ 2 (Japanese standard JIS G4805), which is comparable to AISI L1, and is considered a tool steel. All FMCRD components meet all related codes and standards.

4. It is normal U. S. Practice to use Ferrite Number (FN) to define the ferrite content of weld filler material. The American Welding Society Standard for Calibration of ferrite measuring instruments (AWS A4.2, 1991) specifically

notes that Ferrite Number should not be used for castings and refers to ASTM A-800 for determination of ferrite in castings. ASTM A-800 very clearly defines casting ferrite content in per cent. The inappropriate use of ferrite number for castings in the ABWR SSAR was recognized during the Lungmen NPS OSAR preparation and was corrected accordingly in PSAR Section 4.5.2.4. The per cent limits for casting are identified in the PSAR in accordance with the GE's normal practice.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-014, Item 3

PSAR Sections: 4.5, Reactor Materials

Question Date: December 2, 1997

PSAR Question:

3. Please explain the difference of the materials of the peripheral fuel supports between ASME SA479 Type 316L (non-tube shape) described in the PSAR (4.5.2) and the ASME SA312 Grade Type-304L or 316L (tube shape) described in the GE SSAR. Please explain why the NDE inspection method for wrought seamless tubular products needs to be implemented in peripheral fuel supports in spite of their non-tube shape?

Response:

3. Previous BWR peripheral fuel supports (PFS) were machined from tubular materials made from ASME SA312 Grade Type-304L or 316L. Therefore they were examined per ASME Section III, NG-2500, or more specifically per NG-2550 for tubular fittings and products.

   However, the Lungmen ABWR PFS (as well as the PFS installed in the currently operating ABWRs), are machined from the ASME SA479 Type 316L material which is a bar material. For NDE, Lungmen PFS will receive the required examination per ASME Code, NG-2500, or more specifically per NG-2540. Since the ABWR PFS are not a tubular fitting per NG-2541(b), they need not be examined per NG-2550.

   The reason for the change to ASME SA-479 bar material is that the manufacturing of the PFS was more suitable from the bar as the starting material. The producibility of the ABWR PFS was improved because the welds are eliminated from the PFS design. As can be seen on the attached PFS product drawing, 233C4008 (Revision 1, not issued, under review, GE PROPRIETARY), zone c-2, the peripheral fuel support piece flow orifice is fixed (a drilled hole) and integral to the entire PFS assembly. On the other hand, the standard BWR PFS, starts out as a fabrication from a tubular material, next bayonet pins (3) are welded into the PFS, and finally a separate orifice assembly is installed into the entire PFS and held into place by the bayonet pins. Thus for the Lungmen design, the PFS is a one piece assembly with no fabrication welds. The table below summarizes the differences between the bar and tubular materials:

Material	ASME SA479 Type 316L bar stock material	ASME SA312 Type 316L tubular (pipe) material
Chemical Composition (maximum)	C - 0.030 (required* $\leq$0.020) Mn - 2.00 P - 0.045 S - 0.030 Si - 1.00	C - 0.035 Mn - 2.00 P - 0.040 S - 0.030 Si - 0.75 Cr 16.00 - 18.00

	Cr 16.00 - 18.00 Ni 10.00 - 14.00 Mo 2.00 - 3.00 N - 0.06 - 0.10* Co - 0.050 (required* <0.030)	Ni 10.00 - 15.00 Mo 2.00 - 3.00
Mechanical Properties	Mechanical tests - 1. Properties at room temperature as required by base material 2. Properties at 288°C Min. Yield Stress 107 MPa (ASME Appendix I Table I-1.2) 3. Properties at 288°C Min. Yield Stress 133 MPa* Hardness test - < Rockwell B-92 per ASTM A370 NDE - NG-2540	Mechanical tests - 1. Properties at room temperature as required by base material 2. Properties at 288°C Min. Yield Stress 107 MPa (ASME Appendix I Table I-1.2) Hardness test - < Rockwell B-92 per ASTM A370 NDE - NG-2550
Heat Treatment	1038°C - 1149°C time at temperature 15 min. / 25.4 mm and liquid quench	1038°C - 1149°C time at temperature 15 min. / 25.4 mm and liquid quench
Environmental Considerations	Manufactured for and used in ABWR and BWR environments	Manufactured for and used in ABWR and BWR environments

* Requirements per Table A (Suffixes X, G, K) of Reactor Internal Components Material Specification, 24A5896, Rev.1, referenced on Lungmen Drawing 233C4008. The table (GE PROPRIETARY) is attached with this response. Also, it can be seen from the table that the Lungmen requirements for the SA-312 and SA-479 materials are same using same suffixes G and K.

Further, the discussion in PSAR subsection 4.5.2.3 needs to be revised to reflect that the manufacturing of the CRD housing is not the same as that of the PFS, because the PFS material was changed. Also, the reference in the discussion to the examination requirements of NG-5000 needs to be deleted, because NG-5000, which covers NDE of welds and weld edge preparations, does not apply to weldless ABWR PFS and CRD housing manufacturing.
Accordingly PSAR Subsection 4.5.2.3 will be revised as follows:

4.5.2.3 Non-Destructive Examination of Wrought Stainless Steel Seamless Tubular or Bar Products

The stainless steel CRD housings (CRDHs), which are partially core support structures (inside the reactor vessel), serve as the reactor coolant pressure boundary outside the reactor vessel. The CRD housing material is supplied in accordance with ASME Section III Class 1 requirements. The CRDHs are examined and hydrostatically tested to the ASME Section III Class 1 requirements as well as Class CS requirements.

The peripheral fuel supports are supplied in accordance with ASME Section III, Class CS requirements. The material is procured and examined by ultrasonic methods according to Paragraph NG-2500.

Wrought seamless tubular products for other internals are supplied in accordance with the applicable ASTM or ASME material specifications These specifications require a hydrostatic test on each length of tubing.

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

Track Number: **04-014, Item 3, ROC-AEC Comment Faxed May 4, 1998 by TPC**

PSAR Sections: **4.5, Reactor Materials**

Question Date: **December 2, 1997**

PSAR Question:

3. (**Original Question:** Please explain the difference of the materials of the peripheral fuel supports between ASME SA479 Type 316L (non-tube shape) described in the PSAR (4.5.2) and the ASME SA312 Grade Type-304L or 316L (tube shape) described in the GE SSAR. Please explain why the NDE inspection method for wrought seamless tubular products needs to be implemented in peripheral fuel supports in spite of their non-tube shape?)

   ROC-AEC Comment (second time) on GE's Response Revised to address the ROC-AEC Comment First Time:

   "Paragraph 4.5.2.3, Section 2 (line 8) to be revised to read as "The material is procured and examined by ultrasonic methods according to Paragraph NG-2500, or more specifically per NG-2540" is recommended."

   ROCAEC is still concerned about the examination code adopted by PFS. NG-2550 item (d) requires that Wrought seamless tubular products and fittings (including flanges and fittings machined from forgings and bars) greater than $^3/_8$ inch thickness shall be ultrasonically or radiographically examined in accordance with NG-2552 or NG-2553. AEC consider that the previously attached drawing for PFS machined from bar with something like flange looking seems falling into this regime of jurisdiction and PFS shall be examined per NG-2550 rather than NG-2540. Please clarify."

Response:

3. (**Original Response/Revision:** Original response with GE's Letter GEAE-1998-0200, April 7, 1998 was followed by an e-mail revision of April 21, 1998 to address the first ROC-AEC comment. The e-mailed revised response and this response are included in the file attached with Letter GEAE-1998-0291, dated May 13, 1998.)

   Response to Above ROC-AEC Comment:

   An ASME III Code consultant outside of GE has reviewed the ROC-AEC comment and concluded as follows:

   The ROC-AEC Review comment is in regards to examination of the SA-479 bar material used in the reactor vessel core support structure. The ROC-AEC comment raises the question of whether the peripheral fuel supports, which are made from SA-479 bar, should be examined in accordance with the provisions of NG-2540 (Examination and Repair of Forgings and Bars) or NG-2550 (Examination and Repair of Seamless and Welded Tubular products and Fittings). All of the requirements for examination in various subarticles of NG-2500 are based on the product form identified in the material specification. For example NG 2540 applies to the material procured per SA-479, "Specification for Stainless Steel Bars and Shapes for Use in Boilers and Other Pressure Vessels," because the scope statement in SA-479 clearly identifies the material as bar. Therefore, Section III requires the examination requirements of NG-2540, straight beam ultrasonic examination, to be met. If NG-2550 were used instead of NG-2540, the Code requirements would not be met, and the final assembled core support structure could not be Code Stamped. It should be noted that Paragraph NG-2551(d), which is referenced by NG-2541(a) and quoted by the ROC-AEC, does not apply, because PFS as a core support structure material is not functionally a pressure retaining "seamless tubular product or fitting," nor is it made to the requirements of a tubular material. Further, Paragraph NG-2541(b) does not apply, because PFS is not a "forged flange or fitting."

Some of the reasons behind the Code requirements are as follows:

1. ASME Code requirements for examination of products are based on the material manufacturing process used to make the product form for the material specification. Thus, the examination requirements for plate, forgings, and castings are all different. For example, even though a casting may be bored to resemble seamless tube, it is required to be examined as a casting, regardless of the fact that it will be bored at a later time. Also, the Material Manufacturer is required to examine the item in accordance with the material specification and Section III for the product form.

2. If the product were made to the requirements of a tube material specification, it would be examined to the Section III examination requirements for tubular products. This is because the examination requirements in the Code are based on the forging process for forgings, the extrusion process for tube, and the casting process for castings. Often the strength of the material is based on the manufacturing process and thickness for the product as manufactured. Plate material often has higher strength than castings made to the same chemistry and heat treatment.

3. Because the core support structure is a structural product, the use of bar material instead of tubular product material seems appropriate. In this application, the product is not performing any significant pressure retaining function.

In accordance with the comment, the previously proposed revision to PSAR Subsection 4.5.2.3 will be modified as follows:

4.5.2.3 Non-Destructive Examination of Wrought Stainless Steel Seamless Tubular or Bar Products

The stainless steel CRD housings (CRDHs), which are partially core support structures (inside the reactor vessel), serve as the reactor coolant pressure boundary outside the reactor vessel. The CRD housing material is supplied in accordance with ASME Section III Class 1 requirements. The CRDHs are examined and hydrostatically tested to the ASME Section III Class 1 requirements as well as Class CS requirements.

The peripheral fuel supports are supplied in accordance with ASME Section III, Class CS requirements. The material is procured and examined by ultrasonic methods according to Paragraph NG-2540.

Wrought seamless tubular products for other internals are supplied in accordance with the applicable ASTM or ASME material specifications These specifications require a hydrostatic test on each length of tubing.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-015

PSAR Sections: 4.6.1.2 Description (CRD)

Question Date: December 2, 1997

PSAR Question:

Regarding the CRD separation detection, please provide the possible failure mode of the two separation reed switches. Are those two reed switches designed to fail safe?

PSAR Response:

There are two independent Class 1E reed switches used for detection of the FMCRD separation condition. The design of the FMCRD is such that when both the weight of the control blade and the hollow piston are resting on the ball nut of the FMCRD, this weight compresses an associated spring sufficiently (within the separation detecting equipment within the FMCRD spool piece) to cause a magnet to become positioned sufficiently close to the reed switches so that the reed switches will be closed and will easily conduct current (i.e. are in low impedance state). If either the weight of the control blade or the combined weight of the control blade and hollow piston are lifted off the ball nut, the spring will expand causing the magnet to move away from the reed switches far enough so that the lowered magnetic field strength at the reed switch will result in the reed switches opening (i.e. high impedance state) and thus cannot conduct current easily. The associated Essential Multiplexing System equipment senses the open/close status of the reed switches based upon detection of the high or low impedance conditions. Separation switch A status condition of each FMCRD is sensed by Division I Essential Multiplexing System Equipment. Separation switch B status signals is sensed by Division II Essential Multiplexing System Equipment. Associated separation status are also transmitted to the non-safety Rod Control and Information System (RCIS) for activation of the associated individual FMCRD rod block function, if the separation condition is detected based upon either of the associated reed switch status signals associated with a FMCRD.

After initial installation (and after performing subsequent FMCRD maintenance that required a mechanical uncoupling and recoupling of the control blade to the FMCRD), a coupling check operation is performed. During this operation, the control blade backseats and a mechanical separation condition of the FMCRD will occur, so the operability of both separation switches can be confirmed during this uncoupling check operation.

Thereafter, during reactor operation, the fail safe features of this design depend on the failure scenarios as described below:

- For the case of no mechanical separation condition existing, if a reed switch fails open, this will be treated the same as FMCRD separation and the associated alarm and rod block functions will be activated. For this particular failure case, the design is indeed fail safe. However, for the same mechanical case of no separation condition existing and one of the reed switch failing in the closed status, there is no problem if indeed there is no mechanical separation. If the independent reed switch is still operable, FMCRD separation detection capability still exists. If both reed switches fail closed, FMCRD separation detection capability would be lost. However, in the very unlikely situation that both Class 1E switches fail and if the hollow piston should separate from the ball nut, the redundant spring-loaded latch fingers in the hollow piston will prevent the control rod drop from occuring.

- If the FMCRD indeed becomes mechanically separated, if either one of the FMCRD switches is operable, the alarm and rod block functions will work. If either one of the reed switches fails open, this is also treated as a mechanical separation condition (and thus fail safe). Only if both reed switches have failed closed, the FMCRD separation would not be detected. But, as mentioned above, this is a very unlikely scenario involving multiple independent failures.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-016

PSAR Sections: Ch 4

Question Date: November 27, 1997

PSAR Question:

It is claimed that Lungmen plant has better capability for load-following:

1. How is its load-following characteristics better than existing BWR's ?

2. What is the range for load-following as daily load cycles, load regulation, frequency control and contingency operation?

3. Do the power-flow map and Chapter 15's analysis cover the above load-following operation?

PSAR Response:

1. The Lungmen ABWR will have the GE12 core design with 10x10 lattice array, part-length rods, and an additional eighth spacer. The principal benefit of this fuel design is lower kw/ft rating during operation, with accompanying improved reliability and improved neutron efficiency. The GE12 fuel has improved MCPR and stability performance along with the PCI resistant barrier fuel design, with no PCIOMR operations limitation. The GE12 ABWR Lungmen design will accommodate control blade exchanges throughout the cycle. This design results in the availability of additional control cells at all times during the cycle that can be used for additional flexibility during load following. Besides an improved fuel design the Lungmen design provides use of both automatic control rod movement and automatic core flow changes to provide for improved operability to support automatic load following. Conventional BWRs only provide manual movement of control rods. Lungmen can accommodate an operating flow window of 85% to 111% of rated core flow at 100% power.

2. The ABWR is designed for a 24-hour daily load following cycle with the following profile: starting at 100% power, power ramps down to 50% power in two hours, power remains at 50% for two to ten hours, and then ramps up to 100% in two hours. Power remains at 100% for the remainder of the 24 - hour cycle. In the daily load following mode, the ABWR uses both automatic recirculation flow control and automatic control rod movement to control the reactor power level. The Lungmen design will also accommodate load following down to 25% of rated power which might be used for load reductions below 50% on weekends, for example.

The ABWR is also designed to automatically respond to grid frequency changes. In this operating mode, the ABWR will satisfy peak-to-peak power change demands of 10% of rated power at 5% of rated power per minute when core flow control can be used. Frequency control will be provided while performing ramp power changes required for load following within the power operating range of approximately 65 to 100% of rated power along the rated rod line (the exact range depends upon the control rod line being used for frequency control operation.) The ABWR is designed for frequency control operation throughout the operating life of the plant. Thirty-five peak-to-peak swings per day of operation are permissible.

3. The Automatic Power Regulator (APR) provides for automatic control of either generator or reactor power by changing control rod positions and/or core flow during normal operation. The power range mode of APR operation allows the operator to specify a desired main generator or reactor thermal power and have APR automatically adjust core flow and/or rod position (following a predefined power/flow map trajectory) to accomplish the desired results. The APR follows predefined power/flow map trajectories (and associated restrictions) with the Lungmen control systems, assure that the reactor is not operated beyond its licensed limits for normal power / flow map operation. The APR design will support the contractual requirements for load following and maneuvering in response 2 above. The APR does not attempt to control the plant during upsets, major plant transients (e.g., turbine trip or load rejection) or scrams, so the PSAR Chapter 15 analyses are consistent with load following operation.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-017

PSAR Sections: Ch 4

Question Date: January 5, 1998

PSAR Question:

Steam separator limit line is one of the boundaries of the power/flow operating map. This line is different from that of the standard BWR/6 design. Please explain why there exists such a difference. Please also provide detailed information on how that line is established.

PSAR Response:

The BWR/6 power/flow map has a lower boundary that is set to prevent cavitation damage to the jet pumps and the recirculation flow control valves. This boundary is not required for the ABWR plant because the ABWR plant is designed to use reactor internal pumps and does not utilize the BWR/6-type jet pump and recirculation flow control valve components.

However, the ABWR power/flow map has a less restrictive boundary based on the steam separator-dryer performance limitation. This boundary is identified as the "Steam Separator Limit" line. The Limit Line on the operating map is based on an evaluation of the data from separator-dryer performance tests of three steam separators and a proportionate dryer section. The Limit Line represents the restriction that was determined to be appropriate to assure that moisture content in the steam leaving the vessel does not exceed 0.1 weight %. The placement of the line takes the following items into account: (1) operation up to the high water level alarm (Level 7), (2) performance data from separator-dryer testing, and (3) performance data from BWR plant operation (observed to be better than corresponding data from separator-dryer prototype testing). The Limit Line has been confirmed as appropriate based on actual startup test data of measured moisture content of steam leaving the reactor vessel for the lead ABWR plant.

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

ROCAEC Review Comments :

Please clarify if there are any cavitation requirements on RIP.

Further Clarification

The RIP Net Positive Suction Head (NPSH) requirements are shown in PSAR Table 5.4-1. In actuality if a RIP cavitation limit line was placed on PSAR Figures 4.4-1 and 4.4-2, it would correspond to the graphs' horizontal x- axis (percent core flow) at 0 percent power. The RIP pump suction is physically low in the reactor pressure vessel annulus at approximately 2068 mm above reactor bottom head invert. A RIP runback signal is generated at low steam flow and high core flow conditions where excessive water carry over can occur in the steam separators.

Additional Comments Faxed July 27, 1998:

1. What is the real meaning of setting RIP cavitation limit line on the horizontal x-axis (0% core power) of Figures 4.4-1 and 4.4-2 of Lungmen PSAR? Does it mean no requirement on RIP cavitation limit? So, please state more clearly whether there is RIP cavitation limit requirement are not.

2. Would you please show where it is described on Lungmen PSAR for RIP runback on low steam flow and high core flow conditions?

Additional Further Clarifications per TPC Fax:

1. As can be seen from Table 5.4-1, there is more than adequate NPSH available for the RIPs operating rated conditions. Therefore there is no cavitation interlock restriction on the Power to Flow Map. Were it necessary to show some representation of the Reactor Power Level vs Recirculation Flow, it would correspond to a line superimposed on the X- Axis. Minimum Power Limitation requirements at high flow conditions are imposed based on Steam Separator performance (moisture carryover) requirements.

2. To prevent excessive carryover under low steam flow and high core flow conditions, a steam separator flow limiter restriction is required (shown as Region II on Figures 4.4-1 and 4.4-2. The Recirculation Flow Control (RFC) System logic compares reactor power (from the Neutron Monitoring System) to validated core flow to determine if operation on the P/F map is acceptable or if operation on the P/F map is unacceptable. If operation on the P/F map is unacceptable a RIP runback signal is generated. The closest description of this event is found in PSAR Subsection 7.7.1.3(6) Abnormal Conditions.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-018

PSAR Sections: 4.2 (4.2.1.1 Fuel Assembly)

Question Date: October 17, 1997

PSAR Question:

In Section 4.2.1.1 Fuel Assembly, please add the GE design basis for fuel assembly under a safe shutdown earthquake (SSE).

PSAR Response:

PSAR Section 4.2.1.1 will be revised by adding Item (4) as follows:

(4) Incore loading predicted to occur from a safe shutdown earthquake combined with loss of coolant accident loadings during faulted conditions as discussed in Subsection 3.9.1.4.8.

(Note: Please see response to Question 04-020, Part 2, where Subsection 3.9.1.4.8 is also mentioned regarding update of PSAR Reference 3.9-2.)

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-019

PSAR Sections: 4.2 Fuel Design (4.2.2.2 Control Rods)

Question Date: October 17, 1997

PSAR Question:

1. Please provide the description (including material used) and drawing to show where the bottom casting and the four U-shape absorber tubes are located in the control rod assembly, and how will they be connected to other parts to form a rigid structure.

2. Why there are no holes in the lower transition piece? Please mark clearly which portion in Figure 4.2-2 in active poison length.

3. What were the past performance experiences with this control rod? Are there any design or fabrication related failures?

PSAR Response:

1. Figure 4.2-2 is accurate in showing the structure of the Control Rod assembly. Paragraph 4.2.2.2 has some errors in the description which resulted in the question. In the second paragraph, "...U-shaped absorber tubes.", must be corrected to read, "...U-shaped sheaths which surround the absorber tubes." The remainder of the section explains that these pieces are welded together.

   Second paragraph of Section 4.2.2.2 Control Rods to be revised as follows;

   4.2.2.2 Control Rods

   The control rod assembly consists of a sheathed cruciform array of either stainless steel tubes filled with boron carbide powder or solid hafnium rods. The main structure of a control rod consist of a stainless steel top handle, a stainless steel bottom casting and stainless steel control rod drive coupling, a stainless steel vertical center post, and four stainless steel U-shaped sheaths which surround the absorber tubes. The top handle, bottom casting and center post are welded into a single skeletal structure. The U-shaped sheaths are welded to the center post, handle and castings to form a rigid housing to contain the absorber tubes. Rollers at the top handle and bottom casting of the control rod, guide the control rod as it is inserted and withdrawn from the core.
2. The lower transition piece of Figure 4.2-2 is the same as the "bottom casting" of paragraph 4.2.2.2. This piece (as shown) is a solid stainless steel casting, which contains no absorber material, and requires no cooling holes. The PSAR Figure 4.2-2 will be revised to show the active absorber length (approximately from the bottom row to top row of flow holes).

3. Other than the current operating ABWRs, past performance experience of the control rods comes from BWR-2/6 control rods, which have identical absorber sections (Duralife D-120 and Duralife D-230). GE provided 180 of 205 Control Rods for each of two operating ABWRs (per scope of contract with remaining Control Rods supplied by another vendor), with no reported problems/failures within the GE scope of supply. One operating ABWR just completed its first refueling outage (1/98).

   There have been over 800 D-230 and over 100 D-120 control rods delivered to BWRs between 1987 and 1998. There have been no identified failures of current design D-120 or D-230 control blades in BWR-2/6. The original design D-120 control blades experienced creviced corrosion cracking in the outermost two absorber tubes. This was documented in GE Service Information Letter (SIL) No. 579. The blade sheath hole pattern was subsequently modified to minimize any creviced condition and the rated nuclear life was reduced to prevent future absorber tube failure. The D-120 control rod lifetime is limited to 34% depletion to mitigate the potential for Crevice Corrosion Cracking of the outermost absorber tubes as described in GE SIL No. 579. The D-230 control rods, which have hafnium in place of the outermost tubes as well as in the blade tip, have a much increased lifetime over the D-120.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-020

PSAR Sections: **4.2 Fuel System Design**

Question Date: October 17, 1997

PSAR Question:

1. Since Taiwan is very prone to seismic events, please make a comparison of the seismic resistance of GE12 with that of GE9B which is currently being used in Chinshan NPP.

2. What will be the seismic design procedures to obtain the fuel response? Please provide the topical report if a generic approach has been taken by GE.

PSAR Response:

1. The GE9 and GE12 fuel bundles have both been evaluated to assure that they can withstand horizontal seismic loads which might occur from horizontal seismic acceleration of 3.9g. The Chinshan GE9 fuel uses 100 mil uniform thickness fuel channels and the Lungmen GE12 fuel design uses channels with 120 mil thickness corners and 75 mil thickness side walls. The GE12 fuel channel has been evaluated to assure capability to withstand a seismic acceleration of 3.1g and the GE9 uniform thickness 100 mil channel has been evaluated to assure a capability of 3.6g.

2. The detailed description of the seismic analysis methods are described generically in report NEDE 21175-3-P-A, "BWR Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of Coolant -Accident (LOCA) Loadings (Amendment 3)," October 1984. A copy of this report is attached for your reference. The earlier version of this report, PSAR Reference 3.9-2, is referenced in PSAR Sections 3.9.1.4.8 and 3.9.2.2.4. Reference 3.9-2 will be updated to reflect Amendment 3.

(Note: Please see response to Question 04-018, where Subsection 3.9.1.4.8 is also mentioned regarding update of PSAR Reference 3.9-2.)

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-021

PSAR Sections: 4.5.1 Control Rod Drive Structural Materials

Question Date: November 20, 1997

PSAR Question:

1. In subsection 4.5.1.1(2), the tensile strength of 17-4 PH (265.3 MPa) is too low when compared with the yield property. Please explain and provide with a correct value.

2. Section 4.5.1.1(1)(a), by what will the drive shaft be hardsurfaced?

3. Section 4.5.1.1(1)(b), is the specification of separation magnet as Alnico 5? Please clarify.

PSAR Response:

1. The tensile strength of ASTM A-564, TP 630 (17-4) will be revised to 965.3 MPa, along with the elongation of 14% minimum. The last sentence of subsection 4.5.1.1(2), Special Materials will be revised to read as follows:

   The ball screw shaft and ballnut are ASTM A-564, TP 630 (17-4PH))(or its equivalent) in condition H-1100 (aged 4 hours at 593°C), with a tensile strength of 965.3 MPa minimum, yield of 792.9 MPa minimum, and elongation of 14% minimum.

2. The Drive Shaft will be hardened with Stellite No.6 and Meteco No.15E. Section 4.5.1.1(1)(a) Drive Shaft to be revised to read as follows:

Drive Shaft	ASTM 479 Grade XM-19 (Hardsurfaced with Stellite No. 6 and Meteco No. 15E)

3. The separation magnet will be Alnico 5 or equivalent. Alnico 5 or equivalent is specified for the PSAR for flexibility in the final design. Section 4.5.1.1(1)(b) Separation Magnet to be revised to read as follows:

Separation Magnet	Alnico 5 or equivalent

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-022

PSAR Sections: 4.5.2 Reactor Internal Materials

Question Date: November 20, 1997

PSAR Question:

1. To avoid stress corrosion cracking (scc), low carbon grade stainless steel (c<0.03%) is applied in ABWR SSAR for the materials contact with coolant at temperature greater than 93°C. In section 4.5.2.1, the carbon content of most materials is 0.03% (which is not <0.03%), some are even greater than 0.03%, such as SA182 grade F316L (0.035%), SA-479 type XM-19 (0.06%), and SA-312 grade TPXM-19 (0.06%), what is the designer position on this discrepancy?

2. In section 4.5.2.4, the delta ferrite content of weld metal is 20 FN maximum in ABWR SSAR as opposed to 13 FN maximum in PSAR. Please explain the difference. What version of RG 1.31 was applied?

3. In section 4.5.2.5, the annealing temperature of 2,000°C was applied for Alloy X-750 in ABWR SSAR, it is 1093°C in stead in Lungmen PSAR. Please explain.

4. In section 4.5.2.5, alloy stellite 6 is used as the hardsurfacing material for HPCF Couplings. It is realized that stellite 6 is Cobalt-base alloy, a major radiation contributor. Would it be a concern in this application? What will be the drawback of using a Cobalt-free material as the alternative.

PSAR Response:

1. In accordance with PSAR Subsection 4.5.1.2, GE requires that for Lungmen reactor internal materials that the carbon content for the 300 series stainless steel to be ≤ 0.02% for temperature ≥ 93° C. Therefore, the Lungmen design specification imposes a more stringent requirement of the carbon content for these steels than those specified by the ASME code. The carbon content for Grade XM-19 stainless steel is similarly required to be ≤ 0.04%. So there is no discrepancy because the vendors of these components will follow the material requirements of the design specification.

2. The RG 1.31, Rev. 3, which is applied to the Lungmen reactor internals, recommends the ferrite content in the weld metal to have a maximum ferrite number (FN) of 20. However, both the Lungmen bid specification and the material specification require the ferrite content be 13 FN maximum. This is a more stringent control of the ferrite content to offset excessive dilution of the weld metal.

3. The annealing temperature of 2000°C shown in the ABWR SSAR is a typographical error. It should be 2000°F, which is same as 1093°C as stated in the Lungmen PSAR.

4. Alloy Stellite 6 will not be used as the hardsurfacing material for the HPCF couplings. In section 4.5.2.5, the paragraph on Stellite 6, "Stellite 6 (or its equivalent) hard surfacing ... or plasma arc surfacing process" will be changed to "Hard chromium plating surface is applied to the austenitic stainless steel HPCF couplings."

   GE requires that the chromium plating thickness in the range of 50-100 μm (10^{-6} m). The vendor which supplies the HPCF coupling will provide the plating procedure subject to GE's review and approval. The chromium plating on the HPCF shall be cobalt free. The purpose of the HPCF hard surfacing is mainly to protect the sliding portion of the HPCF coupling due to the differential thermal movements of the shroud and the reactor vessel. The frequency of occurrence for this HPCF coupling movement is much less than that for the guide roller movement inside the control rod drive. Therefore hard chromium plating is more suitably to be applied to the HPCF couplings and not to the control rod drive hard surfaces. Note that, the pins and rollers, described in PSAR Section 5.2.3.2.2, are part of the control rod that can be in the active fuel region. These parts are cobalt free. The guide rollers described in Section 4.5.1.1 are part of the control rod <u>drive</u> system which is located below the core plate in the lower plenum region. These parts are made of Stellite which contains cobalt and these are different pins and rollers from those in the control rod. Therefore, there is no inconsistency in the PSAR.

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

ROCAEC Review Comments :

1. Response accepted. Further clarification is needed on how to require vendor into compliance in the future on the carbon content since the current material specification is different from the design value.

2. Response accepted.

3. Agree with the modification.

4. Agree with the modification.

Further Clarification:

It is a three step process. GE writes an equipment purchase specification that includes material requirements or references a GE material specification. This would include the requirement for 0.020% maximum carbon. The selected vendor then writes a material specification for the material to be used in manufacturing specific components. That vendor's material specification is in turn reviewed by GE for compliance with GE's purchase specification prior to material ordering. Finally, once the material is procured, a GE source inspector reviews that material certification for compliance with GE's original purchase specification.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-023

PSAR Sections: 4.5.2

Question Date: February 3, 1998

PSAR Question:

Alloy 600M was manufactured based on the stabilization parameter N (determined mainly by the contents of Nb and C). It has also been demonstrated that certain minimum value of N needs to be provided for satisfactory resistance to IGSCC or IGA. However, such condition is not fully addressed in the ASME code case N-580. Please clarify what percentage of Nb and C contents will be used for the Shroud Support material of Lungmen design.

PSAR Response:

ASME Code Case N-580 only specifies the requirements on the chemical composition, heat treatment and the mechanical properties of the Alloy 600 with Niobium (Columbium) addition. It does not however address the alloy requirement related to IGSCC.

Both the ASME Code Case N-580 and the Lungmen material specification require the Niobium content of the alloy shall be 1% minimum and 3% maximum, and the Carbon content shall not exceed 0.05%. To improve the IGSCC property of Alloy 600, GE uses the stabilization parameter (N) greater than or equal to 12 as a guide controlling Niobium to Carbon (Nb/C) weight ratio.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-024

PSAR Sections: Ch. 4

Question Date: May 4, 1998

PSAR Question:

Please provide the following documents:

1. Reference 4.3-5: NEDO-32505P, Nov. 1995
2. Reference 4.3-6: NEDE-32417P, Dec. 1994
3. Reference 4.4-3: NEDO-10299A, Oct. 1976
4. Reference 4.4-7: NEDO-31960 & Sup. 1
5. NEDO-32047, Feb. 1992
6. NEDO-32164, Dec. 1992
7. Final Report for the LaSalle FMCRD In-Plant Test Program, Oct. 1989

Response:

The requested references are transmitted as the attachment to this response except for item 5 and 6. These two items, which are not referenced in the PSAR, are ATWS-related documents which were prepared for the BWR Owners' Group and have very restricted distribution and therefore won't be able to provide for ROC-AEC's reference.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-025

PSAR Sections: Ch. 4 Sec. 5

Question Date: May 2, 1998

PSAR Question:

1. In the SRP 4.5.1.II.1 material specifications, the maximum yield strength of cold-worked austenitic stainless steel for CRD materials was specified to be 90,000 psi but in PSAR 4.5.1.1 it was not specified. Please clarify.

2. SRP section 4.5.2.II.2 on acceptance criteria of welding control specified that the core support structure and core internals should conform to ASME Sec. III NG-4000 but in PSAR 4.5.2.2 it was indicated that some core internals do not have to conform to ASME requirements. Please explain.

Response:

1. Cold worked materials are not used in the FMCRDs, and therefore, the yield strength limit for cold worked material was not stated in PSAR Section 4.5. When cold working (bending, straightening, etc.) becomes necessary to the base material in an annealed state, the hardness after cold work is controlled to limit material properties, including yield strength, in the range required for annealed material.

2. "Other internals," in PSAR Section 4.5.2.2 means internals other than the core support structures (CSS). These are either safety-related or non-safety related internals with no core support function. For example, such safety-related (non-CSS) internals are core differential pressure lines and incore guide tubes, and the non-safety related internals are shroud head and steam dryer. There are no formal ASME III Code requirements for these other internals to conform to. The materials and controls on welding for these internals conform to the applicable industry codes and standards, such as ASTM, ASME or AWS. The materials are discussed in PSAR Section 4.5.2.1.

The PSAR Section 4.5.2.2, Controls on Welding, will be revised as follows:

The questioned sentence in PSAR Section 4.5.2.2, "Other internals are not required to meet ASME Code requirements" will be changed as follows:

"The internals, other than the core support structures, will meet the requirements of the industry standards, e.g., ASME or AWS, as applicable."

ROCAEC Review Comments:

1. The response states that ""Other internals," in PSAR Section 4.5.2.2 means internals, such as the shroud head and steam dryer, that are not core support structures (and are not safety related) and therefore, do not have to conform to ASME requirements." What are the other components besides shroud head and steam dryer that are "other internals" ? Provide a list of such internal components which do not have to conform to ASME requirements ?

2. SRP 4.5.2.II.2 specified that BOTH core support structures AND reactor internals have to conform to ASME requirements. But PSAR stated that "Other internals are not required to meet ASME Code Requirements" which is in direct conflict with SRP. Please explain why those non-safety related internals (shroud head and steam dryer) do not have to conform to ASME requirements ?

Further Clarifications:

1. In view of the ROCAEC comments, the original response to Part 2 of the question is revised above. The meaning of "Other internals" in context of PSAR Section 4.5.2.2 discussion is now clarified in the above revised response. These internals are identified in Section 3.9.5.1(2), and some of these, that are non-safety related internals, are identified by an asterisk (*) on Pages 3.9-57 and 3.9-58. (The non safety-related components are shown as follows: Shroud head * and steam separators assembly*, Steam dryers assembly*, Internal pump differential pressure lines*, and Surveillance sample holders*).

   As the revised response indicates, there are no formal ASME III Code requirements for the internals listed in Section 3.9.5.1(2) to conform to. The materials and controls on welding for these internals conform to the applicable industry codes and standards, such as ASTM, ASME or AWS. This is reflected in the revised response above and the revised PSAR wording included with it.

2. The following shows that SRP 4.5.2 is concerned only with the safety-related internals. As discussed above, the core support structures conform to the ASME Code requirements; and the other safety-related and non-safety related internals conform to the applicable industry codes and standards, because there are no ASME Code requirements for them.

SRP 4.5.2 starts as follows:

I. AREAS OF REVIEW

Section 50.55a, "Codes and Standards," and General Design Criterion 1, Appendix A, 10 CFR Part 50 requires that structures, systems, and components important to safety shall be designed, fabricated, and tested to quality standards commensurate with the importance of the safety function to be performed. The purpose of this SRP section is to review and evaluate the adequacy of the materials selected for the construction of the reactor internal and core support structures, and to assure that these regulations are met for those structures.

The ABWR FSER (NUREG-1503), Section 4.5.2 starts as follows interpreting above requirement:

The acceptance criteria used as the bases for the staff's evaluation of reactor internal materials are SRP Section 4.5.2. The reactor internals are acceptable if the design, fabrication, and testing of the materials used in the reactor internal and core support structures meet the code and standards commensurate with the safety function to be performed so that the relevant requirements of GDC 1 and Section 50.55a of Title 10 of the Code of Federal Regulations, Part 50 are met.

The SRP 4.5.2.II.2 requirements are to be interpreted in context of the regulations Section 50.55a, "Codes and Standards," and General Design Criterion 1, Appendix A, 10 CFR Part 50, which limits the scope of SRP 4.5.2 to internals that are important to safety. As the FSER mentions the codes and standards to be applied should be commensurate with the safety function to be performed.

At the end of FSER Section 4.5.2, the following is noted: "*The staff finds that the information in the SSAR related to reactor internal materials meets the criteria of SRP Section 4.5.2 and is, therefore, acceptable.*"

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-026

PSAR Sections: 4.6

Question Date: November 21, 1997

PSAR Question:

Please explain that the Hydraulic Control Units would have enough flow and pressure to push the control blades to the scram position under any anticipated operating (AOO) and accident conditions.

Response:

By design, each CRD Hydraulic Control Unit (HCU), provides sufficient stored energy to scram two CRDs at reactor pressure. Additionally, each Fine Motion Control Rod Drive (FMCRD) mechanism initiates electric motor-driven insertion of its control rod simultaneously with the initiation of hydraulic scram. This provides a diverse means to assure control rod insertion. The HCU accumulator is precharged with nitrogen with the charging water header providing final accumulator charging pressure. Safety-related pressure instrumentation is provided in the charging water header and if charging water header pressure degrades, control rod block and /or reactor scram occurs to assure the safe shutdown of the reactor. Also, the charging water header contains a check valve and a downstream accumulator. This single accumulator is sized to maintain the charging water header pressure downstream of the charging header check valve above the scram setpoint until the standby CRD pump starts automatically.

The HCUs would have enough flow and pressure to push the control blades to the scram position under any anticipated operating (AOO) and accident conditions, based on the above design information.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-027

PSAR Sections: 4.4.3

Question Date: April 29, 1998

PSAR Question:

1. In comparison the Fig. 4.1-1 and Fig. 4.4-2, there is a slight difference about the steam separator limit line. It passes 100% flow at 40% power in Fig. 4.4-2, and at 43% power in Fig. 4.4-1. Please explain the difference.

2. Regarding to Fig. 4.4-1, why the present pump speed evaluated only up to 99% instead of 100%? Please explain.

Response:

1. The equation for the steam separator limit line on the power-flow operating map is as follows:

   CF = 1.08P + 56 at Normal Water Level

   where: CF = Core Flow, Percent of Rated (56.2×10^6 kg/hr)
   P = Reactor Power, Percent of Rated (3926 Mwt)

   At 100% Core Flow, the corresponding Power should be 40.7%. The two power to flow map PSAR Figures 4.4-1 and 4.4-2 will be redrawn for consistency.

2. The maximum permitted core flow during normal operation is 111% of rated. This core flow is attained with ten RIPs at 99% pump speed. In Figure 4.4-2, the maximum core flow attainable with nine RIPs at 100% pump speed is approximately 103% of rated.

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

Track Number: 04-028

PSAR Sections: 4.2

Question Date: June 8, 1998

PSAR Question:

There is only brief introduction of GE12 fuel design in PSAR. Hence, please provide the following documents:

(1) Licensing criteria for fuel design, NEDO-31908, Jan. 1991.

(2) GE standard application for reactor fuel, (GESTAR-II), NEDE-24011-P-A-10, Feb. 1991.

Response:

(1) Reference provided as attached..

(2) Reference provided as attached, and note that NEDE-24011-P-A-10, Feb. 1991 incorporates changes for the following :

- Incorporation of GE8x8NB-1, -2, and -3 fuel designs,
- Fuel licensing acceptance criteria, and
- Fuel channel bow effect on thermal margins.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-029

PSAR Sections: Ch. 4

Question Date: June 10, 1998

PSAR Question:

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. It seems that the flow resistance of RIP design under tripped condition is higher than that of traditional jet pump design.

1. Can we make the statement that the stability issue will be more severe in Lungmen than BWR-6 plant when the core flow is below minimum flow?
2. Has the phenomenon aforementioned considered in Lungmen stability analysis? If the answer is yes, please describe it in detail.
3. Has the reduced natural circulation flow rate been considered in RHR shutdown cooling operation analysis?

Response:

1. Generally speaking, stability margin would be decreased for lower core flow at the same power level. However, the ABWR design has incorporated many design features to assure that stability performance in the normal operating region (Regions I and IV) is more stable than the current operating BWRs. These design features include:
   (1) Smaller inlet orifices, which increase the inlet single-phase pressure drop.
   (2) Wider control rod pitch, which increases flow area and reduces the void reactivity coefficient.
   (3) More steam separators, which reduce the two-phase pressure drop.
   (4) Automatic logic, which prevents plant operation in the region with the least stability margin (Region III).

2. The Lungmen stability analysis considers the limiting power/flow boundary state points on the Region III boundary (see also response to ROC-AEC Track Number 04-011). Region III is defined as the high-power/low-flow area in which the system is most susceptible to reactor instability. Operation within Region III is precluded by Selected Control Rod Run-In (SCRRI) logic. The instability protection provided by Region III has been analyzed . The upper right corner of Region III (68.7%P/39.2%F) is found to be the most limiting state point (core decay ratio 0.54) . The lower left bottom corner of Region III (30%P/20%F) is less limiting than the upper right corner point.

3. The natural circulation flow rate has been considered in the RHR shutdown cooling analysis during emergency shutdown cooling operation.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-030

PSAR Sections: 4.2

Question Date: October, 17, 1997

PSAR Question:

Please provide the performance experiences (including max. rod power and rod burnup) and failure experiences (including failure statistics and root causes) of the GE12 fuel.

Response:

General Electric Company Proprietary Information

The GE12 fuel design is an evolutionary 10x10 progression from the GE11/GE13 9x9 fuel designs. The GE12 design incorporates many of the fuel rod and assembly design features proven by extensive irradiation experience with the GE11/GE13 fuel designs, such as the use of part-length fuel rods, two large central water rods, thick/thin Thermal Size Anneal channels, etc. For this reason, the GE11 experience base is directly applicable to the GE12 design and is summarized in Table 1. The primary failure causes for the GE11/GE13 fuel designs have been debris fretting (3 of 10), and undetected manufacturing defects (3 of 10), with the remaining leakers not yet inspected to determine the leaker root cause (4 of 10).

It is important to note that the manufacturing-related failures discussed above, occurred in the very early GE11 fabrication period (1992). Shortly thereafter, the root cause of these defects was determined and corrective actions taken to avoid recurrence. Additionally, as a part of the GE Zero Leaker program, a Fabrication Best Practices Task Force was also formed shortly thereafter, and identified a number of Best Practices aimed at improved fabrication quality in areas important to fuel reliability. The effectiveness of the identified fabrication best practices is evident in the more recent reliability performance. Since the introduction of Process 6 fuel rod cladding in 1994, over 975,000 fuel rods have been fabricated and placed in operation. With this population, only 7 fuel rod failures have occurred. Six of these 7 leakers have been confirmed by poolside examination to have failed as a result of debris fretting. The one remaining leaker has not yet received poolside examination to determine the root cause of failure. Therefore, the fabrication quality contribution to the GE Zero Leaker program is clearly approaching it's goal.

The equivalent experience information for the GE12 design is presented in Table 2. It is important to note, as shown in Table 2, that several poolside inspections have been performed on a number of the GE12 Lead Use Assemblies, extending to lead bundle average exposures of ~43 GWd/MTU, with demonstrated excellent fuel performance. No GE12 fuel failures have occurred.

General Electric Company Proprietary Information

Table 1. GE11/GE13 Experience

Plant	Number of Bundles	Number of Fuel Rods	Number of Leakers	Number of Inspections	Bundle Avg. Exposure* GWd/MTU	Comment
LUAs						
USA-A	4	296	0	1	13.0	In 2nd cycle
USA-B	4	296	0	3	37.6	In 4th cycle
EUR-A	8	592	0	4	35.0	In 5th cycle
USA-C	4	296	0	2	35.0	Discharged
EUR-B	8	592	0	5	41.0	In 6th cycle
USA-A	4	296	0	3	26.8	Discharged
USA-D	4	296	0	1	38.0	Discharged
USA-E	4	296	0	2	31.0	Discharged
USA-F	4	296	0	1	34.3	Discharged
USA-G	4	296	0	0	42.0	Discharged
EUR-C	4	296	0	3	42.0	Discharged
EUR-D	2	148	0	3	48.0	In 7th cycle
USA-H	2	148	0	3	43.6	Discharged
Reloads						
62 reloads	11,312	837,088	10	6	52.0	

Table 2. GE12 Experience

Plant	No. of Bundles	No. of Fuel Rods	BOL	No. of Leakers	No. of Inspect	Bund. Avg. GWd/MTU	Comment
LUAs							
EUR-A	2	184	11/97	0	0	0.0	In 1st cycle
EUR-E	4	368	7/97	0	0	0.0	In 1st cycle
EUR-F	8	736	4/97	0	1	11.0	In 2nd cycle
Chinshan 1	4	368	10/96	0	1	12.0	In 2nd cycle
EUR-A	4	368	7/96	0	0	10.6	In 2nd cycle
USA-I	4	368	5/96	0	0	13.0	In 2nd cycle
USA-B	4	368	4/96	0	1	14.0	In 2nd cycle
USA-J	4	368	10/95	0	1	12.0	In 2nd cycle
EUR-C	4	368	11/94	0	2	27.0	In 3rd cycle
EUR-A	4	368	8/94	0	3	23.0	In 4th cycle
EUR-G	8	736	8/93	0	5	43.0	4 in 6th cycle
Reloads							
EUR-F	60	5520	5/98	0	0	0.0	In 1st cycle
EUR-G	120	11040	3/98	0	0	0.0	In 1st cycle
EUR-A	128	11776	11/97	0	0	0.0	In 1st cycle
USA-K	252	23184	11/97	0	0	0.0	In 1st cycle
USA-C	192	17664	12/96	0	0	0.0	In 1st cycle
EUR-A	146	13432	7/96	0	0	10.6	In 2nd cycle

* At the end of the last completed cycle

USA- USA Plant EUR - European Plant

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-031

PSAR Sections: Ch. 4

Question Date: June 19, 1998

PSAR Question:

1. Compare Fig.4A-5a and 4A-6a, with core flow not changed, cycle exposure increases from 1.1 GWD/MTU to 2.2 GWD/MTU, control rod location did not change. Is this reasonable ? Please explain.

2. Compare figure 4A-8b and 4A-9a, the flow rate did not change, cycle exposure increases from 4.4GWD/MTU to 5.5MWD/MTU, Control rod was not withdrawn, instead was inserted from notch 12 to notch 10. Is this reasonable ? Please explain.

Response:

1. Control rods are adjusted to maintain a critical condition in the reactor. During the cycle, the core reactivity is affected by the burnup of gadolinia and the depletion of U-235. During some periods of operation, these two factors compensate and the total core reactivity does not change. Under these conditions, the control rod pattern can be maintained constant. The simulation provided in Figure 4A-5a and 4A-6a indicates that the core reactivity is relatively constant during the period from 1.1 to 2.2 GWd/MTU and therefore minimal adjustments are required to the control rod pattern. The control rod patterns shown in the figures are based on simulations of typical conditions in which the core exposure is incremented and all core conditions are assumed to remain constant during that increment (e.g., core flow, power, control rod patterns). During actual operation, minor adjustments to core flow or control rod positions may be necessary to maintain the desired power level.

2. See also the response in item 1 above. During some periods of operation, it is possible that the reactivity increase associated with the gadolinia burnup is greater than the reactivity decrease associated with the depletion of U-235 and therefore the total core reactivity increases. This will require additional control rod insertion to maintain the reactor in a critical configuration.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-032

PSAR Sections: Ch. 4.4

Question Date: June, 22, 1998

PSAR Question:

The following questions in regard to the operation of SCRRI system:

1. It can be seen from Figure 4.4-1 that the right boundary of the region III is around 40% flow. However, according to the response to question 04-011, SCRRI is bypassed if the flow is greater than 36%. Shouldn't SCRRI unbypassed region cover the region III? Please explain?
2. Is there any normal operation condition, such as start-up or partial power operation, that core flow reduction due to inadvertent single RIP trip may cause the reactor operating within region III domain?
3. When the reactor is under 9 RIP operation, will it still require two or more RIPs tripped in order to put SCRRI in auto mode?
4. Reconsider question (2) for the condition of 9 RIP operation.

Response:

1. The 36% core flow represents the analytical limit for the functions of control rod withdrawal block and selected control rod run-in to be bypassed. PSAR Figures 4.4-1 and 4.4-2 will be revised, so that Region III will be bounded by the 36% constant flow line, maximum rod line, 30% constant power line, and the natural circulation line. SCRRI will not be bypassed in Region III.

2. An inadvertent single RIP trip during startup will not cause flow to drop lower than the 36% core flow analytical limit, therefore, will not cause the reactor to operate in Region III.

3. The SCCRI algorithm uses the inputs of 2 or more out of 10 RIPs (ASDs) not normal (tripped or manually shutdown) with core flow less than or equal to 36% (Analytical Limit) and reactor power greater than or equal to 30% (Analytical Limit). Therefore if 9 RIPs are in operation and another RIP trips for a total of 2 RIPs not operating normally, with the above power and core flow conditions, the SCRRI auto initiates.

4. The core flow for 8 RIP operation with minimum speed is approximately 36%. Therefore, if one pump is inadvertently tripped when the reactor is under 9 RIP operation, it will not enter into Region III.

PSAR Figures 4.4.1 and 4.4.2 will be revised as described above in response number 1.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 04-033

PSAR Sections: 4.4

Question Date: June 20, 1998

PSAR Question:

The following questions in regards to 9 RIP's operation.

1. Is the core flow distribution under 9 RIP's operation the same as 10 RIP's operation? If there is any related ABWR flow test information, please provide.

2. Is there any reverse flow through the tripped RIP when the reactor is under 9 RIP's operation? If there is, what is the amount of reverse flow?

3. Is the CPR calculation applicable with 10 RIP's operation the same as 9RIP's operation?

Response:

1. The pressure differential across the lower plenum is insignificant compared to the pressure differential between the lower and upper plenum. Therefore, the flow distribution under 9 RIPs operation will be the same as 10 RIPs operation. GE has conducted measurements for jet pump BWRs under single loop operation which demonstrates this.

   No core flow distribution anomaly has been observed at the two currently operating ABWR with 9 of 10 RIPs in operation.

2. There is reverse flow through the tripped RIP. With the reactor at 100% core flow, but with 9 RIPs in service, the maximum reverse flow through the tripped RIP is expected to be -5,370 m^3/hr. See also response to ROC-AEC question 15-026 item #2 on similar subject.

3. Yes, the CPR calculation for 10 RIP operation is applicable to 9 RIP operation. The core flow is derived from differential pressure across the core plate or pump deck differential pressure common to all RIPs, so the core flow determination is unaffected by the idle RIP.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-001

PSAR Sections: Ch 5.2

Question Date: December 2, 1997

PSAR Question:

In Table 5.2-4, the specification listed for Ni-Cr-Fe alloy steel of RPV Nozzles and Stub Tubes is Code Case N-580. It was found that said code has not been approved by NRC yet. Please explain if it is feasible.

PSAR Response:

This Code Case was issued by the ASME Code on May 26, 1997. The NRC regularly updates approval of Materials Code Cases by revising Regulatory Guide 1.85. The last revision (Rev. 30) of Regulatory Guide 1.85 was published in October 1994, per Regulatory Guide List dated October 31, 1997. Therefore, eventually, the Code Case N-580 is expected to be included by the NRC in the Regulatory Guide 1.85 as all the NRC's represenatives on Section II, Section III, Section IX and the Main Committee of the ASME voted positively for the Code Case. As 10CFR 50.55a specifies, code cases not included in Regulatory Guide 1.85 need to be approved by the Director of Nuclear Reactor Regulation. TPC is seeking this approval with the ROC AEC Director of Nuclear Reactor Regulation. The ROC AEC has received information on this code case and has jurisdiction over Lungmen. The NRC has had no applications for use of this code case by plants located in the U.S.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-002

PSAR Sections: 5.4.13

Question Date: January 14, 1998

PSAR Question:

Please explain when SRV solenoid valves will be operated by pressurized air? Please explain why each SRV, designated for automatic depressurization purpose, is equipped with another individual air accumulator instead of nitrogen accumulator?

PSAR Response:

Please note that the safety relief valves referred to in PSAR Section 5.4.13 are for the Control Rod Drive System (C12), Standby Liquid Control System (C41), Residual Heat Removal System (E11), High Pressure Core Flooder System (E22), Reactor Core Injection System (E51), and Reactor Water Cleanup /System (G31). As for the safety/relief valves (SRVs) for the Reactor Coolant Pressure Boundary (RCPB) overpressure protection, please refer to PSAR Section 5.2.2.

The following reply is prepared for the safety relief valves (SRVs) in the Main Steam System (B21) for RCPB overpressure protection function:

Whenever the primary containment is inerted with nitrogen, the instrument air piping serving equipment inside the primary containment will be supplied with nitrogen from the nitrogen supply (N2)system rather than instruemt air.

During normal plant operation, the SRV solenoid valves will be operated by pressurized nitrogen and not by pressurized air. Nitrogen is used to operate the SRVs because the SRVs are located inside the containment building, and the containment atmosphere is inert with nitrogen gas. Also, the exhaust gas from the SRV pneumatic system is not allowed to affect the containment inert atmosphere. The nitrogen supply to the SRV is sufficiently reliable to preclude loss of availability from the supply source. And, for the Automatic Depressurization System (ADS) designated SRVs, the nitrogen supply is backed up by its stored supply when the low nitrogen pressure event occurs. The only time when the SRV solenoid valves will be operated by pressurized air is when the containment building is open, and when nitrogen gas is not available during plant shutdown.

It is not correct that each SRV designated for automatic depressurization purpose is equipped with another individual air accumulator instead of nitrogen accumulator. For each dual function, direct-acting type SRV, the valve function as a spring-loaded safety valve for overpressure safety operation. And a nitrogen accumulator is supplied for the overpressure pressure relief function. As for the ADS SRVs, they function as spring-loaded safety valves for overpressure safety operation. In addition, one nitrogen accumulator is supplied for the pressure relief function, and another individual nitrogen accumulator is supplied for the ADS function. Hence, the ADS SRVs are not supplied with any air accumulator.

Further Clarification to ROCAEC's Comments:

The only time the SRV solenoid valves will be operated by pressurized air is when the primary containment vessel (PCV) is open or when nitrogen gas is not available during plant shutdown.

The last sentence on PSAR page 5.2-7 will be revised to read "Each SRV selected for automatic depressurization is equipped with an additional pneumatic accumulator..." instead of "with another individual air accumulator...".

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-003

PSAR Sections: 5.4.7.3.3

Question Date: January 14, 1998

PSAR Question:

In section 5.4.7.3.3, it is described that RHR in the shutdown cooling mode should be capable of bringing the reactor to cold shutdown conditions (100 °C)...... The temperature requirement is not consistent with that defined in Table 16.1.1-1 (i.e., ≤ 93 °C). Explain the difference.

PSAR Response:

The value 100 °C in section 5.4.7.3.3 in the PSAR was inadvertently incorrect. The correct definition of cold shutdown was established as ≤ 93 °C as stated in the Technical Specifications of the PSAR, Table 16.1.1-1. The definition of Cold Shutdown as ≤ 93 °C is consistent with previous plants such as BWR/6 Technical Specifications. Section 5.4.7.3.3 will be corrected. This error also appears in Sections 5.4.1.1.7, 5.4.1.1.8, 5.4.7.3.3, 7.4.1.1 and 15.2.9. These sections will also be corrected.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-004

PSAR Sections: 5.2.4

Question Date: November 26, 1997

PSAR Question:

5.2.4.7 Code exemptions (P 5.2-32)

Last paragraph : These portions of systems are specifically identified in Table 5.2-8.

Review Comments :

(1) The above information was not found in the said Table. Please supply the information required.

(2) Please incorporate the following considerations into the code exemptions

1. Exempt only from examination, not including testing.
2. Relief requests shall be granted by AEC.
3. The relief requests shall be reviewed periodically, if new techniques, equipment or changes in the plant have made what was once an impossible examination now a possible examination, the relief requests will be revoked.

PSAR Response:

(1) Exemptions from the ASME Section XI requirements for specific system components are identified under the "Sec. XI Exam Cat." Column of Table 5.2-8. For example, all piping 25 mm and smaller for systems E11, E22, E51, G31 and N22 are shown to be "Exempted per IWB-1220 (b) (1)". The phrase shown for E11 will be revised to read "Exempted per IWB-1220 (b) (1)" instead of "Exempted per IWB-1220 (2) (1)".

(2)

1. Only exemptions from examinations per IWB-1220 are described in

Section 5.2.4.7. There is no discussion of testing in Section 5.2.4.7 and it is therefore inherent that there are no testing exemptions. No changes will be made to Section 5.2.4.7.

2&3. The following sentences will be added to Section 5.2.4.7 to accommodate relief requests:

"The inservice inspection program for Class 1 components and any inservice inspection program relief requests will be reviewed based on the Code Edition in effect and inservice inspection techniques available immediately prior to the time of application. Each approved inspection program relief shall be reviewed periodically to ensure that the relief is still applicable regardless of any changes in inservice inspection techniques, equipment or accessibility."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-005

PSAR Sections: Ch 5

Question Date: November 19, 1997

PSAR Question:

1. The second paragraph of Section 5.2.2.4 in page 5.2-8 explained that the SRV discharge lines in the wetwell will meet the ASME B&PV code section III, class 3 piping requirements but in Note h of Table 3.2-1d of Chapter 3, the non-destructive testing of the said piping has upgraded it to class 2 piping. Please clarify.
2. Please confirm that the last paragraph of Section 5.2.2.4 in page 5.2-8 which reads "The accumulator capacity is sufficient for one actuation at drywell design pressure or five actuations at normal drywell pressure whichever is not limiting", the word "not limiting" should really be "limiting" ?
3. The second paragraph from the last in Section 5.2.2.10 reads "This includes setpoint, operating time capacity and blowdown requirements." What is the meaning of "operating time capacity" ? Please explain.

PSAR Response:

1. The second paragraph of Section 5.2.2.4 (page 5.2-8) will be changed to be consistent with Note h of Table 3.2-1d of Chapter 3. The sentence in the second paragraph of Section 5.2.2.4 will be changed to "The SRV discharge lines in the wetwell air space are classified as Quality Group C and Seismic Category I, all welds shall be non-destructively examined to the requirements for ASME B&PV Code, Section III, Class 2 piping".

2. The phrase "not limiting" will be changed to "limiting".

3. The phrase "operating time capacity" has no meaning, it will be changed to "operating capacity".

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-006

PSAR Sections: 5.2.3

Question Date: November 21, 1997

PSAR Question:

The following inconsistencies were identified after compared with the ABWR SSAR, please explain each of them:

1. In 5.2.3.2.2, the corrosion potential is 50 to 100 mV_{SHE}, it was 50 to -50 in ABWR.
2. In 5.2.3.2.2, the conductivity control is less than 0.1 mS/cm (<0.2 mS/cm in ABWR)
3. In 5.2.3.2.2. 3(4), dissolve oxygen is controlled to 15 ~ 50 ppb, but it was 20 ~ 50 ppb in ABWR design.
4. (a) In the second paragraph of 5.2.3.4.1.1, the low carbon (0.03%) grade material was not specified.
   (b) The nuclear grade material was applied restrictedly on application where the solution heat treatment is not applicable.
5. In 5.2.3.4.2.1, the delta ferrite content of 8 to 13 is requested, but it was 5 to 20 in ABWR SSAR.
6. The specification of water chemistry (Table 5.2-5) has included the values of pH and ECP in ABWR SSAR, but were not covered in Lungmen PSAR. The concentrations of the impurities requested in Lungmen PSAR were generally lower than what described in SSAR, but the oxygen of the condensate and hot standby reactor water is greater instead (200 ppb ® 300 ppb).
7. The value of E should be also defined in 5.2.3.2.2.4 when the fluence level was specified.

PSAR Response:

1. Since the generation of the ABWR SSAR document, considerably more data have become available for ECP measurements under normal water chemistry conditions, and older data have been more thoroughly reviewed. The current consensus is that a range of +50

to +100 mV (SHE) more adequately describes the range of potentials experienced in BWRs under normal water chemistry conditions.

2. Since the generation of the ABWR SSAR document, there are additional data to suggest that even with Hydrogen Water Chemistry (HWC) operation, there is still evidence of crack initiation and crack propagation when the routine reactor water conductivity is 0.2 uS/cm. At the same level of HWC, the incidence of cracking is significantly reduced when the reactor water conductivity is reduced to 0.1 uS/cm. One of GE's goals for optimum water chemistry is a reactor water conductivity of less than 0.08 uS/cm.

3. Since the generation of the ABWR SSAR document, a more recent version of the EPRI BWR Water Chemistry Guidelines was developed with participation from GE. At the last guidelines meeting, there was no hardened evidence that operation with a feedwater oxygen concentration of 15 ppb would lead to accelerated corrosion of carbon steel, relative to operation at 20 ppb. Accordingly, since there was reluctance from the BWR community to endure supplemental feedwater oxygenation when the benefits of doing so were questionable, the lower limit of 15 ppb was endorsed.

4. (a&b)
For welded components exposed to service temperature at or above 93 °C, the carbon is limited to 0.02% which is less than (more restrictive) that the limit of 0.03% specified in the ABWR SSAR. However, if the component is subsequently solution heat treated after welding, the as-welded condition does not exist anymore; and therefore, the above carbon control is not necessary. As a general practice, high carbon (up to 0.08%) materials are not used for any welded components if they are to be left in the as-welded condition. The only exception to the practice of controlling carbon to ≤ 0.02% are for specialized components where fabrication or operating conditions demand an alternative material. Specific examples are control rod absorber tubes and in-core monitor components. Absorber tubes will be made of a special high purity, tantalum

stabilized version of type 304 called 304S for improved IASCC resistance. No weld filler metal is used on such thin-walled parts. See response to Item 1 of ROC-AEC Question Track Number 04-022 for additional details on specific material usages.

5. The ABWR SSAR allows ferrite content of 5 to 20. However, TPC Bid Specification limits this requirement to 8 to 13. Our plan is to meet the TPC requirement, which is in compliance with the ABWR SSAR requirement (i.e. 8 - 13 range is well within the 5 - 20 range).

6. The measurement of ECP under normal water chemistry conditions does not have much significance. Since the Lungmen Project will not initially operate with HWC, the parameter was removed from the table. Along with the many ambiguities of the measurement, such as where the measurement is made and at what feedwater hydrogen concentration, there was additional incentive to remove it from the table. For operation with HWC, the guideline is still a potential reading of less than -230 mV (SHE).

   With a reactor water conductivity of 0.1 uS/cm, the theoretical range of pH is extremely limited ~6.6-7.6. At this conductivity there are no materials related chemistry concerns if the coolant is slightly acidic or slightly basic. With water of this purity, the accurate measurement of pH is generally not observed without very sophisticated techniques. Accordingly, since the measurement is of limited utility, it was also removed from the table.

   The overall lower chemistry limits were approved for use on the Lungmen Project based on the need to meet the requirements of the more recent version of the EPRI BWR Water Chemistry Guidelines. The oxygen limits for reactor water are only applicable for increasing the water temperature above 60 °C (140 °F) during startup periods.

7. We agree to clarify by adding "($E_f > 1$ MeV)".

ROCAEC Review Comment:

4. Further clarification is needed on the requirement of carbon content of the austenitic stainless steel of the non-weld components.

Further Clarification:

4. One of the key methods of controlling IGSCC of austenitic stainless steel in BWRs is avoidance of use of material in a sensitized condition. Stainless steel that has been subjected to solution heat treatment and rapid quenching is completely devoid of sensitization. This is the normal condition for all stainless steel raw materials. However, subsequent heating by fabrication processes such as hot forming or welding can cause the stainless steel to become sensitized. There are several ways to avoid or eliminate sensitization in the final structure: (1) Assemble the structure without welding so the material remains in the solution heat treated condition; (2) Control carbon to a low level, e.g. 0.020%, so the material does not sensitize when welded; (3) Re-solution heat treatment after welding. GE standard practice for the past 20 years permits up to 0.050% carbon for Type 304 or 316 material, provided it is assembled without welding or is resolution heat treated subsequent to welding. Many studies have been performed to confirm resolution heat treatment eliminates weld sensitization and therefore, the potential for IGSCC. Solution heat treatment has been specifically identified as a method to reduce or eliminate IGSCC for BWR conditions[(1)]. In addition, solution heat treatment has been identified as a remedy for IGSCC, as an alternate to other methods such as using nuclear grade material (i.e., 0.020% carbon)[(2)]. The factor of improvement in IGSCC resistance of standard (0.06%) carbon material by solution heat treatment has been determined to be 91.1[(3)], which significantly exceeds the threshold criterium for mitigation of a factor of 20. The U.S. NRC has also recognized that solution heat treatment is a remedy for sensitization equal to the use of low carbon material[(4)].

Based on these considerations and data, the GE limit of 0.050% carbon for assemblies in the solution heat treated condition is considered justified and appropriate for Lungmen.

REFERENCES

(1) EPRI NP-944, "Studies on AISI Type-304 Stainless Steel Piping Weldments for Use in BWR Application," September 1978.
(2) EPRI NP-3684-SR, Volume 1, "Proceedings: Second Seminar on Countermeasures for Pipe Cracking in BWRs," September 1984.
(3) EPRI NP-3684-SR, Volume 2, "Proceedings: Second Seminar on Countermeasures for Pipe Cracking in BWRs," September 1984.
(4) NUREG-0313, Rev. 2, 1988.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-007

PSAR Sections: 5.2.4

Question Date: November 24, 1997

PSAR Question:

Section 5.2.4.3.2.1 is on the ultrasonic examination of RPV, the statement "However, the inservice inspection program for the reactor pressure vessel and any inservice inspection program relief request will be reviewed based on the Code Edition in effect and inservice inspection techniques available at the time of application." did not define clearly the responsible organization for the ISI program or the review organization of the Relief Request. Please also clarify for what kind of papers/approvals "the time of application" refers to ?

PSAR Response:

This section comes from the SSAR and is based on the requirements and primary authority of 10CFR50, so a detailed definition of the responsible organization was not included since the regulation on which it is based assigns that responsibility to the Licensee. In this case, the responsible Licensee for this inservice inspection program is TPC. The secondary authority is the ASME Code, Section XI, which assigns responsibility for the inservice inspection program to the Owner, also TPC in this case. The "time of application" refers to the application for a commercial operating license (COL), also derived from 10CFR50, since it is necessary to know the date of commercial operation in order to define the Edition and Addenda of ASME Section XI which will be used for the inservice inspection program.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-008

PSAR Sections: 5.3.2

Question Date: November, 1997

PSAR Questions:

1. In section 5.3.2.1, the statement "ABWR temperature limits are based on a more recent revision of RG. 1.99" Please describe the "more recent" revision of referred RG 1.99.
2. Please explain the procedure of constructing the P-T limiting curve.
3. Please describe the method and procedure for Pre-service and In-service hydrostatic and leak pressure tests.
4. Please provide the initial and EOL (60 years) RT_{NDT} of beltline metal, weld and HAZ at vessel surface, ¼ t, ¾ t, as well as the limiting RT_{NDT} for heatup and cooldown P-T curve generation.
5. Were the measuring uncertainties of temperature and pressure considered in the P-T limiting curve (Fig. 5.3-1)?

PSAR Responses:

1. The RG 1.99, Revision 2 [1] provides the current correlation for calculation of ART (adjusted reference temperature). The revision 1 correlation for ART is based on fluence, % Cu, and % P, while the revision 2 correlation is based on fluence, % Cu, and % Ni.

   REFERENCE:
   1)"Radiation Embrittlement of Reactor Vessel Materials," USNRC Regulatory Guide 1.99, Revision 2, May 1988.

2. P-T curves will be developed using methods that are similar to those provided in Sections 3.0,4.0, and 5.0 of Reference 2. A plant specific description of the methods will be provided when the new P-T curves are provided.

   REFERENCE:
   2) T. A. Caine, "Pressure-Temperature Curve Methods for the

Chinshan and Kuosheng Nuclear Power Stations," GENE, San Jose, CA, April 1994, (GENE-523-167-1193).

3. This information will be provided in the FSAR.

4 GE will provide a set of P-T curves for the heatup and cooldown operating condition at EOL. These curves apply for both the 1/4T and 3/4T locations. When combining pressure and thermal stresses, it is usually necessary to evaluate stresses at the 1/4T location (inside surface flaw) and the 3/4T location (outside surface flaw). This is because the thermal gradient tensile stress of interest is in the inner wall during cooldown and is in the outer wall during heatup. However, as a conservative simplification, the thermal gradient stress at the 1/4T is assumed to be tensile for both heatup and cooldown. This results in the approach of applying the maximum tensile stress at the 1/4T location. This approach causes no operational difficulties, since the BWR is at steam saturation conditions during normal operation, well above the heatup/cooldown curve limits.

Therefore, since the P-T curves are developed only for the 1/4T location, RT_{NDT} values are only calculated for the 1/4T location. For the HAZ material there is no regulatory requirement to provide the initial RT_{NDT} and there is no clear technical guidance for determining the initial RT_{NDT} for the HAZ materials. Therefore, the Initial RT_{NDT} for the HAZ will not be provided. Preliminary calculations of RT_{NDT} are based on the RPV material composition as defined in the PSAR and are provided below:

% Cu	% Ni Range	Initial RT_{NDT} (℃)	ART** (℃)
WELD	**MATERIAL** *		
0.08	0.8 - 1.2	-20	-7.7
BASE	**MATERIAL**		
0.05	0.2 - 1.2	-20	-11.5

* The weld material is the weld adjacent to the beltline

** ART - Adjusted Reference Temperature

Actual material RT_{NDT}'s will be provided when all the CMTR's (Certified Material Test Reports) are available. The actual material RT_{NDT}'s will be used to generate the P-T curves.

5. No. The metal temperature is specified on the P-T curves with no uncertainties included for temperature or pressure. The utility/plant needs to determine what instrumentation uncertainties apply and take those into consideration when using the P-T curves during plant operation.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-009

PSAR Sections: 5.4.7

Question Date: November 19, 1997

PSAR Question:

Please provide the NPSH available versus the NPSH required as a function of RHR flow rate and temperature. Is NPSH required a single value or a function of fluid temperature?

PSAR Response:

The NPSH required is strictly a function of the pump inlet design. For a given pump inlet design the NPSH required for the pump varies as a function of flow.The NPSH available for a system varies as a function of flow and temperature. . NPSH available decreases with increasing flow and temperature and NPSH required increases with increasing flow. Final selection of the RHR pump and specifically the pump impeller and inlet are ongoing. Therefore, the NPSH available at the maximum service temperature and NPSH required as a function of flow will be provided in the FSAR after the pump design details are finalized.

ROCAEC Review Comments:

1. Head is usually expressed in water height. Head can be obtained by dividing pressure from specific gravity. Since specific gravity is different under different temperatures, please clarify why NPSH required is a constant value ?

2. The NPSH required of Centrifugal pump is usually determined by close to constant temperature water. Please clarify what water temperature was used to determine the NPSH required for the ECCS pump ? constant temperature ? or the suppression pool temperature after LOCA ?

Further Clarifications:

1. Both the NPSH Available (NPSHA) and NPSH Required (NPSHR) will vary with water temperature. The minimum NPSHA value is specified in Section 5.4.7.2.2(1) as 2.4 meters. This value is based on the hottest water temperature in the suppression pool, 100 °C. To clarify the information provided in Section 5.4.7.2.2(1), the title for NPSH will be revised to read "Minimum net positive suction head available (NPSHA) at 1m above the pump floor". NPSHR is not specified in Section 5.4.7.2.2(1) and will be provided by the pump manufacturer after final pump design is completed.

2. The RHR system is designed to provide the specified minimum NPSHA for the RHR pump. The maximum suppression pool water temperature after LOCA, 100 °C, was used to determine the specified minimum NPSHA provided in Section 5.4.7.2.2(1). The RHR pump will be designed to ensure that the NPSHR for the pump will be less than the NPSHA over the range of pump operating conditions.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-010

PSAR Sections: 5.4.7.1.1.10

Question Date: November 19, 1997

PSAR Question:

Please provide the sequence of events which might initiate the use of ACIWA.

PSAR Response:

All AC electrical power from onsite or offsite sources is lost.

All Emergency Core Cooling System (ECCS) pumps are unavailable.

Severe accident has occurred and the purpose of ACIWA is to prevent core damage or, if core damage has occurred, to terminate melt progression.

Manually actuated valves E11-BV-0045C and E11-BV-0046C are opened in order to connect Fire Protection System to RHR loop C and provide water to RHR Low Pressure Flooder piping to inject water into the RPV and the drywell spray sparger.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-011

PSAR Sections: 5.4.8.2

Question Date: November 19, 1997

PSAR Question:

1. The operation of the precoat of F/D will greatly affect the frequency of removal and availability period of F/D. Right now, a one step precoat was adopted for the First NPS (Chinshan) and a two steps precoat was adopted for the Second NPS (Kuosheng) and each one has its advantages and disadvantages. Please explain which precoat operation is adopted for Lungmen and the reason?

2. Please provide the methodology for checking the evenness of F/D precoat.

PSAR Response:

1. One of the following two methods of F/D precoating will be used for the Lungmen F/D systems:

   A. The precoat material which is a predetermined amount of slurry of premixed anion and cation resins is applied to the filter area using an eductor and a precoat pump. About 0. 20 lb. of precoat material per square foot of filter area (or 0.30 kg / m^2) is applied to the filter. The precoat pump circulates the slurry around the F/D until the discharge water is clear. This method is essentially the one step method.

   Advantages: 1. Simpler and easy to perform.
   2. Less solid waste
   3. Less expensive
   4. Longer differential pressure cycle

   B. The other method includes applying a thin layer of filter aid (such as cellulose fiber, diatomaceous earth, etc.) on the filter area followed by a layer of precoat material overlay. The process is similar to item A above except there are two layers of dissimilar materials applied, one on top of the other. This method is essentially the two step method

Advantages:
1. Removes more solids
2. Less chances of resin break through

Disadvantages:
1. Longer precoating time
2. Generates more solid waste
3. More expensive
4. Shorter differential pressure cycle

Either of the two above methods can be adopted for the Lungmen F/D design. It is up to the plant owner to decide how they want to control their F/D precoating operations. However, the one step method (method A above) is the preferred method because of its simplicity and its lower solid waste production. For Lungmen, F/D precoat methods will not affect F/D availability since there are two RWCU 100% F/D units per plant.

2. Evenness of the precoat can be checked using any of the following methods.

   A. Salt Intrusion Test - After the precoat material has been applied to the F/D unit, a small amount of non-corrosive ionic salt material is injected into the F/D inlet. An ion-chromatography analysis of the F/D effluent is then performed to determine the amount of the salt removed by the F/D unit. The percentage of salt removed will then be compared to the F/D performance requirements for dissolved solids removal. Precoat evenness is indicated when the salt removal is at or above the F/D performance requirements. evenness

   B. Precoat evenness can also be indicated by measuring the F/D performance during operation immediately after precoating is completed. This is accomplished by comparing the conductivity levels at the F/D inlet and outlet and calculating the dissolved solids removal efficiency. Again, if the dissolved solids removal efficiency is below the F/D performance requirements then additional precoat should be added.

C. Fiber optic probe fed to a TV is another way to check evenness but may be quite expensive.

Any of the three methods described above can be used to measure precoat evenness within the F/D units. It is up to the plant owner to decide how they want to test for precoat evenness. However, method B is recommended since it can be measured directly using existing equipment and would require the least amount of operator action.

ROCAEC Review Comments:

Two methods were adopted for item 1 and three methods were adopted for item 2. What method TPC plans to adopt ? and the reason ? TPC should explain.

Further Clarifications:

Lungmen will adopt method 1A (single step precoat application) for precoat operations based on the advantages described in the original response above.

For checking precoat evenness, method 2B will be adopted since it will be less expensive to implement and will not require additional operator actions to perform. In addition, the current system design supports this method since there are sample points upstream and downstream of the filter demineralizer equipment to measure the inlet and outlet conductivities which will allow the operator to easily determine the performance (precoat evenness) of each filter demineralizer.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-012

PSAR Sections: **5B: RHR Injection Flow and Heat Capacity Analysis Outlines**

Question Date: November 21, 1997

PSAR Question:

Q_{min} is used to determine the initial quantity of water injection of LPFL and is also an indicator during system depressurization the pressure at which point LPFL will start injecting water into the RPV. But in PSAR, the exact value of Q_{min} was not given. Also, please explain clearly in the PSAR the reserved margin value of the pressure head and its factors of consideration in the performance curve of the Inject flow pump.

PSAR Response:

The Q_{min} is the minimum flow from the pump through the minimum flow bypass line to provide a minimum continuous flow of water through the pump to ensure that the pump is protected. The minimum flow for the pump is currently specified as 148 m^3/hr. This value will be confirmed during finalization of the design details of the pump and will be provided in the FSAR. The actual reserved margin is equal to the actual pump head at minimum flow minus the sum of the system resistances. The total system resistance is equal to the sum of the following:

a. the difference between the reactor and drywell pressure (1.55 Mpa at minimum flow)
b. the static pressure head, Hs, (RPV normal water level minus suppression pool level)
c. the pressure drop through the drywell to wetwell vent, Hv
d. the hydraulic head losses, Hf

The reserved margin will be at least equal to 7% of the required pressure head. The 7% margin is provided to allow for uncertainties and pump wear.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-013

PSAR Sections: 5B: RHR Injection Flow and Heat Capacity Analysis Outlines

Question Date: November 21, 1997

PSAR Question:

The analysis of RHR Heat Exchanger in PSAR was based on the fixed heat removal capability of 370.5 kJ/sec-°C. However, when system is depressurized, water injection will increase, S/P temperature will rise and other auxiliary heat sinks (RBCW, RSW) will all affect the water condition at the inlet of RHR and the "fixed" heat removal capability is therefore hard to maintain. So, please review the assumption and provide more information whether it satisfies the heat removal requirements of DBA to confirm its conservatism.

PSAR Response:

A preliminary containment analysis has been performed using the heat removal capacity of 370.5 kJ/sec-°C and the UHS temperature of 35 °C (95 °F). The calculated peak suppression pool water temperature is below the 97.2 °C (207 °F) maximum bulk suppression pool water temperature limit. The final containment analysis and RHR heat capacity analysis will be completed after the detailed system design has been finalized and the results will be provided in the FSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-014

PSAR Sections: 5.4.9 MS and FW Piping

Question Date: December 19, 1997

PSAR Question:

In section 5.4.9.1 Safety Design Bases, please clarify that why dynamic loads induced by SRV discharge to suppression pool was not included in the safety design bases for main steamlines and feedwater piping.

PSAR Response:

The dynamic loads induced by SRV actuation and discharge to the suppression pool have not been specifically addressed in this section because Appendix 3B of Chapter 3 contains a description of the loads which need to be considered in the analysis of each piping system. Thus, for consistency with the rest of the safety design bases contained in Section 5.4, the main steam and feedwater piping system do not contain a description of all the loads that are analyzed. While the dynamic loads induced by SRV actuation and discharge to the suppression pool is not specifically listed in section 5.4.9.1, the analyses of the main steam and feedwater piping systems include the effect of the SRV loads.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-015

PSAR Sections: 5.3.1

Question Date: February 18, 1998

PSAR Question:

1. This section did not list the Ni content of SA-533-B-1 steel in detail. Please list it clearly.

2. Please explain whether the assigned lead factor of 1.2 - 1.5 is sufficient in PSAR.

PSAR Response:

1. Nickel content was not mentioned specifically because it is an alloying element rather than an impurity, and as such, is defined in the basic material specification, SA-533. For SA-533-B-1 the nickel content specified by ASME is 0.40 to 0.70%.

2. The assigned lead factor of 1.2-1.5 shown in the PSAR was provided in the TPC Bid Specification, Appendix A, Chapter 4, Paragraph 2.3.1.8.1. In GE's proposal it was clarified that a lead factor of 1.2-1.5 will be provided relative to the ¼ t (thickness) location in the reactor vessel wall (Ref. GE Lungmen Proposal, Chapter 3-Technical Description, Paragraph 3.3.2.4.4). This is considered to be the most appropriate definition since the ¼ t location is used for calculation of pressure-temperature curves, etc. A factor of 1.2-1.5 will provide more than adequate early warning in the unlikely event that loss of toughness is significantly greater than predicted. The surveillance capsule, as provided, will also comply with ASTM E-185.

   It should be noted that due to the large annulus (between the vessel shell and the core shroud) of the ABWR design, relative to the annulus of older BWRs, the neutron fluence on the ABWR RPV wall is reduced significantly. This, coupled with the much better chemistry of the RPV materials (lower Cu impurities) results in a non limiting maximum Adjusted Reference Temperature (ART) of -11.5°C (11.3°F) at the end of 60 year life. The Lungmen RPVs are predicted to have a shift of 15.4°F or 8.5°C after 60 years. The decrease in the Upper Shelf Energy (USE) is also much smaller for the Lungmen RPVs. Therefore, the Lungmen RPVs have excellent fracture toughness and there is no technical issue which would require obtaining a higher lead factor.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-016

PSAR Sections: 5.4.6

Question Date: February 18, 1998

PSAR Question:

Why there is a big difference between the Design exhaust pressure of 8.62 MPaG in line 4 of Table 5.4-2 in page 5.4-58 and 0.98 MPaG in page 5.4-61 of SSAR? Also, why the System design of the Main flow element of 8.62 MPaG/302 °C in SSAR is replaced by 17.5 MPaG/77 °C ?

Response:

The design pressure of 8.62 MPaG in the Lungmen PSAR is consistent with the value provided SSAR Rev. 9, amendment 37.

The design pressure of the RCIC turbine exhaust line is chosen based on the requirements of both the interfacing system LOCA (ISLOCA) requirements and the ASME Code requirements. Prior to the application of the ISLOCA requirements, the turbine exhaust line was designed for the lower design pressure. Protection, to comply with the ASME Code requirements, was supplied by putting rupture disks in the exhaust line. With the application of ISLOCA requirements, it was necessary to raise the design pressure to a minimum of 2.82 MPaG. If the design pressure of the line were to be 2.82 MPaG, it would still need overpressure protection in compliance with the ASME Code since the reactor operating pressure is 7.17 MPaG. If rupture disks were used to provide overpressure protection, they would need to open at 2.82 MPaG, but this would violate ISLOCA criteria which is to provide burst protection up to reactor operating pressure. Since overpressure protection devices, such as rupture disks could not be used to protect the piping from the reactor pressure, it was decided to design the piping for the reactor pressure. The design pressure which was chosen was the same as the reactor design pressure of 8.62 MPaG.

In the SSAR Rev. 9, amendment 37, the main flow element was combined together with flow restricting orifices. The orifices and the flow element have different requirements and have been separated for the Lungmen PSAR. The higher pressure is correct since the main flow element is located in a pipe segment that is connected to the pump discharge piping. However, as a result of detailed design of the RCIC pump, it has been determined that the actual design pressure of the pump discharge piping is 16.7 MPaG. This value is slightly lower than the previously calculated 17.5 MPaG. Therefore, the system design pressure for the flow element in Table 5.4-2 will be changed to 16.7 MPaG to be consistent with the current pump design.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-017

PSAR Sections: 5.4.6

Question Date: March 6, 1998

PSAR Question:

Page 5.4-18 mentioned that the capacity of the RCIC water supply sources should reference Table 9.2-3 but PSAR Table 9.2-3 has nothing to do with capacity. Please clarify.

Response:

The table reference on page 5.4-18 will be changed from "Table 9.2-3" to "Table 9.2-6". Table 9.2-6 provides the minimum usable volume requirements for the condensate storage tank which is the water supply source for RCIC.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-018

PSAR Sections: 5.3.3.6

Question Date: April 2, 1998

PSAR Question:

In Standard ABWR design, if the coolant temperature difference between the dome and the bottom head drain exceeds 55°C, neither reactor power level nor recirculation pump flow can be increased considering the RPV thermal stress capabilities. But in PSAR this temperature difference has been increased to 80°C. Please explain.

Response:

The Lungmen ABWR RPV design includes a control rod drive (CRD) and incore (IC) penetration design which utilizes stub tubes to join the CRD and incore housings to the bottom head of the RPV. The stub tubes help reduce thermal stresses in these penetrations and allow for a higher operating limit of 80°C, than the 55°C that was used in the BWR/6 design. The BWR/6 RPV had no CRD and IC penetration stub tubes.

The Lungmen RPV design and stress analysis consider the 80°C maximum coolant temperature difference between the dome and the bottom head to increase power or recirculation flow.

Furthermore, it should be noted that Section 7.7.1.3 (5) correctly specifies this temperature difference limit as 80°C, in both the ABWR SSAR and in the Lungmen PSAR.

ROCAEC Review Comment:

The temperature difference was raised from 55 °C to 80 °C which will cause not only increase in thermal stress at the bottom but all the RPV as well. TPC did not provide sufficient explanation. TPC should explain what designs are related to the temperature difference ?

Further Clarification:

The limiting temperature difference of 80 °C as specified in the Lungmen PSAR, Chapter 5, Paragraph 5.3.3.6 is correct. The ABWR SSAR, Chapter 5, Paragraph 5.3.3.6 incorrectly specifies 55 °C as the limiting coolant temperature difference between the top head dome and the bottom head to increase power or recirculation flow. Therefore, the 80°C limit that is specified for the Lungmen reactors does not represent an increase of the limit relative to the ABWR.

As described in the original response above, The Lungmen ABWR RPV design includes a control rod drive (CRD) and incore (IC) penetration design which utilizes stub tubes to join the CRD and incore housings to the bottom head of the RPV. The stub tubes help reduce thermal stresses in these penetrations and allow for a higher operating limit of 80°C, than the 55°C that was used in the BWR/6 design. The BWR/6 RPV had no CRD and IC penetration stub tubes.

The concern associated with increasing flow while there is a large temperature difference between the top and bottom of the RPV, is due to the possibility of the stratification of colder CRD and RIP purge flow in the lower parts of the bottom head dome, while natural circulation may not provide enough mixing at these lower portions of the bottom head. This effect is expected to occur locally at the lower portion of the bottom head and is expected to result in increased stresses in parts that are located only in these lower portions. The stress analysis of parts that are located in this region will evaluate stresses due to this stratified condition. The highest stresses will occur at the CRD and incore stub tubes and in the stub tube-to-bottom head welds. Other parts that are located away from, and above the stratified location will experience slight increases in stress as the colder water in the bottom will mix with the warmer water from the annulus, when the flow is increased, and have a higher temperature as the flow passes away from the bottom head.

ROCAEC Review Comments (8/20/98):

GE's response indicated that the temperature difference could be raised to 80 degrees C from 55 degrees C because of the use of stub tubes. However, stub tubes were used at Chinshan but the temperature difference is still 55 degrees C. How does one explain this?

Further Clarifications (8/24/98):

A review of the Chinshan CRD penetration stress analysis shows that the configuration of the stub tube-to-bottom head connection is of the set-in design, which is different than the Lungmen set-on design. Therefore, it is difficult to make a direct comparison of stresses between the two reactors.

The temperature difference is 80 degrees C for the ABWR RPV. It was demonstrated by stress analysis that this limit is adequate for the existing ABWR. It will also be demonstrated by stress analysis that the 80 degrees C limit is adequate for the Lungmen RPV.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-019

PSAR Sections: 5.2.3.2.2

Question Date: April 2, 1998

PSAR Question:

This section stated that IGSCC resistant materials will be used in Lungmen so hydrogen water chemistry is not needed. But in the same section it was mentioned that this material is susceptible to IGSCC and hydrogen water chemistry will help alleviate this phenomenon. Please explain whether hydrogen water chemistry will be carried out.

PSAR Response:

The wording in the Lungmen PSAR currently includes a discussion of hydrogen water chemistry that was inadvertantly carried over from the ABWR SSAR. Since hydrogen water chemistry is not included in Lungmen, these paragraphs are not appropriate and will be eliminated. The basic point of the paragraph remains the same. That is, all materials for the Lungmen ABWR in contact with high temperature reactor water have been selected for high resistance to sensitization and IGSCC.

ROCAEC Review Comment:

The translation of the question was not correct. (In the following translation the second IGSCC should be IASCC :

This section stated that IGSCC resistant materials will be used in Lungmen so hydrogen water chemistry is not needed. But in the same section it was mentioned that this material is susceptible to IASCC and hydrogen water chemistry will help alleviate this phenomenon. Please explain whether hydrogen water chemistry will be carried out.)

Further Clarification:

The words in the Lungmen PSAR that mentioned IASCC and Hydrogen Water Chemistry were carried over from the SSAR for which HWC was a part of the design. The intent of the pargraph in question was to provide a rationale for inclusion of HWC even though IGSCC resistant materials were being used. Since IASCC is a relatively new phenomenon, the existing field experience and laboratory data are less clear that these materials are also resistant to IASCC. Consequently, for the U.S. standard plant design it was decided to include HWC for an extra margin of resistance to potential cracking phenomenon such as IASCC. However, it is not presently clear that this extra margin is absolutely necessary. All of the known cases of IASCC have occurred in Type 304/304L components. The material used to construct the Lungmen core support structures(316L) is known to be more resistant to IASCC than 304/304L. In general the ABWR core structures reach a significantly lower peak end-of-life fluence than older BWR 4/5s because of improvements in configuration. In addition, measures have been taken to control stresses in the core structures and to locate welds in regions of lower neutron flux. Consequently, it is possible that Lungmen will not be subject to IASCC of the main core structures. On this basis it was concluded that it was not necessary to include HWC in Lungmen at initial startup. IASCC is a time dependent phenomenon that requires a long term buildup of neutron damage. There are a number of operating BWRs that have 316L core structures, some of which have been in service since the mid-1980s. These plants will provide a substantial early warning if IASCC in 316L does become a concern. If, at a later date, it becomes clear that there is some potential for IASCC in Lungmen, HWC protection can be added at that time.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-020

PSAR Sections: 5.4.6

Question Date: April 15, 1998

PSAR Question:

The pipings after the Lungmen RCIC system main pump will go through Oil coolers and Barometric Condenser before it is pumped back to the main pump suction by the Condensate pump (see Fig. 5.4-8). But at Kuosheng, the pipings after Oil coolers will go directly to main pump suction. What is the difference between these two ? The pipings at Lungmen has one more Barometric condenser to go through, wouldn't that introduce another possible leakage path ? Please clarify.

Response:

The design which is shown in Fig. 5.4-8 is the standard ABWR design which is based on the Kashiwazaki 6/7 design. The BWR/6 product line, of which Kuosheng was a part, used a slightly different method of oil cooling and steam leakage control than previous designs such as Chinshan and the standard ABWR. For Lungmen, it was initially decided to use the same design as the standard ABWR and the plants previous to the Kuosheng product line.

Please note: Following the completion of PSAR Fig. 5.4-8, an improved RCIC turbine and pump design is being implemented. This improved design uses process water as a lubricant which replaces the use of oil and eliminates the need for oil coolers. The improved design is simpler and more reliable and substantially reduces the potential for steam leakage. The RCIC turbine and pump improved design will be shown in the revised PSAR Figure 5.4-8 which will be provided in the PSAR Amendment.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 05-021

PSAR Sections: Ch. 5.2 and 7.3

Question Date: May 12, 1998

PSAR Question:

The following statements are given in page 5.2-38 and page 7.3-30 of PSAR respectively: "High flow exceeding the preset value in any of the four main steamlines will result in trip of the MSIV isolation logic to close all the MSIVs"; "High steam flow in any two of the four MSLs will result in trip of the MSIV isolation logic". Please clarify which one is correct.

Response:

We agree that the Chapter 7 text is incorrect. Therefore, the third sentence of Section 7.3.1.1.2(3)(c) [Page 7.3-30, Item (c)] will be corrected as follows:

"High steam flow in any of the four MSLs (as detected by two-out-of-four sensors within each MSL) will result in trip of the MSIV isolation logic..."