審查意見摘要表

（第六章 ～ 第九章）

第六章　特殊安全設施

編號	內　　　　　容
06-01	奧斯田鐵不銹鋼製程控制之規範與取樣
06-02	圍阻體分析數據之澄清(計九題)
06-03	圍阻體冷卻系統是否含圍阻體噴灑模式
06-04	PSI/ISI 規範之澄清(計三題)
06-05	核燃料護套金屬與水之化學反應分析假設之澄清
06-06	RIP 上有機材料之輻射分解及熱解測試
06-07	ECCS 分析程式之適用問題(計二題)
06-08-1	分裂產物移除及控制系統有關問題之澄清(計四題)
06-08-2	控制室適居系統有關問題澄清(計五題)
06-09	圍阻體噴灑系統採取手動起動之妥適性
06-10	本章壓力儀器錶頭採用 SI 單位之妥適性
06-11-1	SRV/ADS 未採 Low-Low Set 功能之澄清
06-11-2	RCIC Rupture Disc 被移除之影響
06-11-3	LPCF 打水必經 RHR 熱交換器之影響
06-12	抑壓池取水濾網堵塞餘裕之假設(計二題)
06-13	圍阻體內真空破壞閥之測試週期
06-14	ACS 之圍阻體隔離閥完全位於圍阻體外側
06-15	圍阻體洩漏測試問題澄清(計三題)
06-16	ACS 之圍阻體隔離信號之澄清
06-17	圍阻體充氮作業之方便性
06-18	可燃氣體控制系統管材之設計資料
06-19	ESF 金屬材料之規範澄清(計三題)
06-20	抑壓池與燃料池之水質規範為何放寬
06-21	圍阻體之高拉力鋼筋是否使用水淬鋼筋
06-22-1	二次圍阻體之負壓值澄清
06-22-2	主蒸汽隧道之 blow-out panel 事故洩放是否排至大氣
06-23	圍阻體隔離系統之有關問題澄(計四題)
06-24-1	如何落實 ANSI N510 之空氣流量分佈測試
06-24-2	如何落實 ANSI N509 中有關活性炭之採購及測試要求

第七章　儀控系統

編號	內　　　　容
07-001	1.要求應承諾IEEE 603而非IEEE 279 2.應依BTP HICB 14提送軟體發展計劃送審 3.RPS PRA分析報告提送
07-002	1.澄清RPS欠缺"主蒸汽管高輻射"及"反應爐高水位"跳脫信號 2.澄清將"High Suppression Pool Temp"納為RPS跳脫之理由 3.RPS擴充性說明
07-003	要求修改各節本文與Table 7.1-2法規承諾總表不一致處
07-004	要求將RG1.153列入章節內，以符合Table 7.1-2
07-005	要求將反應爐各種高度參考點定義清楚並闡明與水位儀器跳脫設定關係
07-006	1.NEMS/EMS對信號如何作最佳取樣，以避免信號失真 2.澄清核四系統、設備組件編碼原則
07-007	NEMS有那些non-safety signals會送到EMS?其界面如何處理?遵循法規為何?
07-008	要求說明EMS及NEMS的故障模式及效應(FMEA)
07-009	與GESSAR相較，NEMS與EMS內容明顯未盡週延。
07-010	要求軟體發展引用之相關法規由草案版改為正式版
07-011	要求依SECY93-087提供核四廠相關評估資料，証明儀控系統之多樣性與深度防禦設計足以抵擋共因失效
07-012	要求提供RPS， SSLC與ESFAS的DD&D分析資料
07-013	要求提供RPS， SSLC及ESFAS軟體發展計劃
07-014	要求提供RPS， SSLC及ESFAS的FMEA
07-015	澄清表7.1-2法規承諾相關事宜
07-016	澄清RPS為何未將主蒸汽管高輻射信號納為跳機信號
07-017	遙控停機盤內是否有數位設備?
07-018	澄清"MSIV閥位信號"與"地震信號"之傳輸方式是Hardwire或EMS?
07-019	請說明Air Header Dump Valve功能測試為何未列於運轉

編號	內　　容
	規範中?
07-020	澄清主汽機stop valve及control valve關閉跳脫之旁通是依主汽機第一級壓力或mode switch位置而定?並修正矛盾之敘述
07-021	依GDC23要求台電提核四廠SSLC的FMEA
07-022	要求說明SSLC軟體如何能符合BTP14有關Consistency與Completeness
07-023	要求說明如何避免操作人員與電腦間產生mode confusion
07-024	要求說明RPS系統未將軟體失效作為跳脫信號之理由(研究用反應器TRIGA有此設計)
07-025	要求針對運轉員Commission error對系統安全功能的影響
07-026	要求依RG1.70 Section7.2提供核四RPS之設計與奇異標準設計及日本K6/K7之比較資料
07-027	要求說明核四數位設計如何防範外來軟體(如病毒、時間炸彈、網路入侵)惡意入侵
07-028	澄清DI&DD是否已將人機互動產生的共通性失效納入考慮?並請詳細說明
07-029	請說明RPS跳脫單元可靠性數據(2.95E-04)如何求得?
07-030	要求說明線上測試法如何能偵測出軟體邏輯錯誤?
07-031	如何確保EMS與NEMS間的“獨立性”
07-032	圖7.8-4顯示PT-025， LT-003與FI-009會造成數位與後備硬接線系統共因失效之虞
07-033	澄清核四有無MSIV手動控制，MSIV狀態硬接線顯示，RPV壓力硬接線顯示等
07-034	7.5PAM欠缺對法規承諾章節
07-035	要求在table 7.5-1增列法規要求文件
07-036	通案檢討第七章與表1.8之不一致處
07-037	RCIS多工器在各種情況下運作詳細資料
07-038	抑壓池水位宜列為Type B而非Type C
07-039	核四PSAR 7.6節內容與SRP 7.6節比較，完整性與週延性不夠
07-040	表7.5-2未對Neutron Flux討論
07-041	EMS與NEMS所用光纖規範相同否?另ANSI ASC x 3T9.5

編號	內　　　　　容
	未列於PSAR表1.8內
07-042	主控制室撤離時機之假設條件合理性
07-043	RSD盤防火區?Transfer SW位置?轉移程序
07-044	RSD盤面部份設備配置不對稱，是否影響停機能力?
07-045	核四PSAR 7.5節內容與SRP 7.5節比較，完整性與週延性不夠
07-046	徐老師提出，對儀控系統作較高層次應注意事項的審查意見，共六項。
07-047	1. NUMAC與SSLC間有無期因失效問題 2.能否手動起動SLC系統 3.EMS失效之肇始事件及事故情節發展
07-048	核四LFCV設計，及其與核二廠LFCV設計優劣比較。
07-049	依審查指導委員會第二分組第二次會議之會議結論提出，該等問題係由清大周教授於會中提出
07-050	共用Sensor之相關問題及失效之影響情形？
07-051	ARI未列於PSAR第7.4節內，理由何在？該功能是否屬安全系統功能

第八章　電力系統

編號	內　　　　容
08-01	Bus A4、B4、C4 之正常電源跳脫後，由後備電源供電 EDG 自動起動否？EDG 可在現場併聯否？達額定電壓頻為何需 20 秒？
08-02	單線圖 8.3-1 與 8.1 章節第 11 頁略有出入
08-03	345KV 輸電線路故障，發電機可否單獨運轉？G/T 為何不做全黑電源？
08-04	FMCDS 是 Class 1E 設備，Mator 為 Non-Class 1E？C3 供給 FMCDS 部份電力為何不 Fast transfer？
08-05	主變、輔變及後備輔變未註明容量？說明 A stuck BKR will not trip more than one additional unit or line？保護電驛使用之型式？補充說明各保護電驛功能？空斷開關使用電動，電氣聯鎖，可手動操作；另加接地開關？開關場是否屋內室？
08-06	澄清 RAT 之隔離問題？變壓器是使用集油槽並排放至安全地區？
08-07	8.3.1.1.1 及 8.3.1.1.7 述 Class 1E 電壓<70%切換是否造成設備損害？
08-08	直流蓄電池是否可用鎳氫電池取代鉛電池？

第九章　輔助系統

編號	內　　　　容
09-01	RBCW 設計溫度與壓力澄清
09-02	最終熱沈的海水溫度設計值澄清
09-03	燃料池襯裡之耐震設計
09-04	輔助燃料池冷卻及補水澄清
09-05	緊急寒水系統熱負載澄清
09-06	大修吊裝設備耐震等級問題
09-07	反應器廠房與輔助廠房用過燃料池之設計差異
09-08	最終熱沈最高溫度限制澄清
09-09	冷凝水補充淨化系統添加物之澄清
09-10	冷凝水補充淨化系統除礦器取樣方式
09-11-1	冷凝水儲存及輸送系統導電度澄清
09-11-2	CST 設計基準之澄清
09-11-3	CST 是否列入安全分析可用系統澄清
09-12	RBCW 為何不考慮 LOPP 說明
09-13	NCW 未包括管閥測試說明
09-14-1	ECW 系統防治水鎚之設計說明
09-14-2	ECW 泵緊急電源穩定性澄清
09-15	寒水系統冷媒相關問題澄清
09-16	RBSW 管路腐蝕問題澄清
09-17	RBSW 及 TBSW 海水處理方式澄清
09-18	PASS 取樣系統經驗缺失改善澄清
09-20-1	消防系統柴油日用槽容量問題說明
09-20-2	消防系統使用國內法規說明
09-20-3	消防系統通風導管問題說明
09-20-4	煙控系統循環情形說明
09-21-1	手提無線電機產生電磁干擾之澄清
09-21-2	通訊系統是否已考慮備份問題說明
09-22-1	照明系統是否應增加備用交流照明設施
09-22-2	安全及非安全區照明設施位置說明
09-22-3	D/G 房間照明設備是否應防爆設計

編號	內　　　容
09-23	緊急直流照明供電時間說明
09-24	RIP M-G set 輸出電壓說明
09-25	一次圍阻體空調系統設計澄清
09-26	燃料池空間設計說明
09-27-1	電纜穿越室火災後停機能力澄清
09-27-2	火警偵檢器裝設位置說明
09-28	燃料池冷卻能力澄清
09-29-1	SLCS 以隔離閥取代爆開閥理由說明
09-29-2	SLCS 改由 HPCF 管路注入後之有效停機能力澄清
09-30-1	火災危險分析(FHA)未提 RBSW 防火措施說明
09-30-2	FHA 評估是否涵蓋兩部機共用設備之影響
09-30-3	一部機運轉，另一部機施工時 FHA 評估方式說明
09-30-4	FHA 評估外在危險因素影響安全停機之準則說明
09-30-5	煙塵移除時產生輻射外釋之可能及影響說明
09-30-6	二次圍阻體煙塵移除方式說明
09-30-7	核四有否使用 Charcoal Canistertype 過濾器說明
09-30-8	安全相關設備免於爆烈物波及之保護方法說明
09-30-9	RHR，RCIC 及圍阻體隔離閥經不同串防火區之電纜保護方法說明
09-30-10	FHA 分析人員資格符合情形說明
09-31	大修吊裝設備是否設有"Disengage"指示燈說明
09-32	儀用空氣隔離閥相關問題澄清(2 項)
09-33	消防系統相關問題澄清(7 項)
09-34-1	ATWS 啟動條件澄清
09-34-2	SLCS 考量 Potential Leakage 說明
09-34-3	SLCS 硼液濃度 25%餘裕是否足夠澄清
09-35	如何防止因不當電氣接地而造成控制室及安全停機室同時發生火災說明
09-36-1	PSW 系統排放標準說明
09-36-2	PSW 系統供水壓力試驗標準澄清
09-37	輔助燃料池相關問題(12 項)
09-38	兩部機組共用設施安全考量
09-39	新燃料接收倉庫澄清

編號	內　　　　　　容
09-40	燃料吊具是否符合相關法規澄清
09-41	消防系統濕式立管引用法規澄清
09-42-1	D/G 潤滑油冷卻水來源澄清
09-42-2	D/G 起動空氣露點說明

審查問題與答覆內容

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-06-001

PSAR Sections: Ch 6.1.1

Question Date: December 31, 1997

PSAR Question:

6.1.1.1.3.3 Austenitic stainless steel annealing & machining: Are there any standards or guidelines during the fabrication/bending process, in regard to controlling of the material range of hardness, surface characteristics ? Is guideline available for reliably obtaining specimen ?

Response:

The guideline used for bending is to limit the bend radius to greater than 20 times the wall thickness of the material unless qualified by testing. If the bend radius is less than 20t, the material hardness is checked in the maximum deformation area. Generally, a process qualification test piece is used, but testing of actual hardware using a portable tester is also allowed. The material hardness must be Rockwell B 90 or less for Type 304/304L material, and Rockwell B 92 or less for Type 316/316L material in the final condition.

Surface finish of the ground stainless steel surfaces that will be exposed to BWR water is controlled by requiring that polishing be performed to a written procedure such that the final finish is achieved using a #120 grit size paper. Such polishing processes have been demonstrated to remove the negative effects of grinding.

No PSAR revision is necessary to reflect the response above. The above details are specified in the Materials and Process Control Specifications as a part of the Purchase Specification Requirements.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **I-06-002**

PSAR Sections: **Ch 6.2.1**

Question Date: **December 31, 1997**

PSAR Question:

1. In section 6.2.1.1.2 it was mentioned that the floor that separates drywell and wetwell is "reinforced concrete diaphragm floor". What is structurally different between this floor and the other floors ?

2. In ABWR Standard Safety Analysis Report (SSAR), the cross section area of drywell connecting vents (DCV) is 1m x 2m while in Section 6.2.1.1.2 it is 1m x 1.8m. Please explain why this change was made and if any impact on the analysis of containment pressure and temperature.

3. In section 6.2.1.1.2, the containment allowable leakage is 0.5% per day but why main steam isolation valve (MSIV) leakage is not included ? Please explain.

4. In 6.2.1.1.3.3.1, for the analysis of feedwater line break, the effective flow area (Fig. 6.2.2) and feedwater flow (Fig. 6.2.3) are not the same as those in SSAR. However, the results are the same. Please explain.

5. Among the references listed for the feedwater line break analysis in 6.2.1.1.3.3.1, References 6.2-1 and 6.2-3 are related to Mark III design. ABWR containment design is different than Mark III design. Do these references affect the ABWR containment analysis ?
6. In 6.2.1.1.3.3.1 (17), the assumption of drywell air flow into wetwell is smaller than the actual case value (Fig. 6.2-5). (Compare actual case value versus model assumption curve narrow area) what is the impact on the results in using smaller flow area ?

8. In the analysis of wetwell-to-drywell negative differential pressure in Section 6.2.1.1.4.1, as far as wetwell to drywell pressure is concerned, using wetwell/drywell spray (see Fig. 6.2-17) will be higher than if wetwell/drywell spray is not used at all (see Fig. 6.2-8) before 75 seconds. This result seems to show that if wetwell/drywell spray is initiated after an accident, the containment pressure will not be lowered and actually increased as a result. Is this reasonable ? Please clarify.

9. Figure 6.2.18 shows that the wetwell-to-reactor building differential pressure is 5.9 kPaD, while SSAR states 0.06 kPaD. Which is correct ?

Response:

1. The diaphragm floor, which separates drywell and wetwell, is equipped with a steel liner on the underside (Subsection 3.8.3.1.1), to provide leak tightness against any direct steam bypass leakage from drywell to wetwell air space. Likewise, the drywell top slab is also equipped with a steel liner to provide leak tightness for the primary containment boundary. Other floors in Reactor Building do not have steel liners.

   No change is required to the PSAR from the response above.

2. There has been no change in the cross section area of drywell connecting vent, the dimension was simply rounded off roughly as 1m x 2m in ABWR SSAR Subsection 6.2.1.1.2.
   No change is required to the PSAR from the response above.

3. The reason for not including the MSIV leakage in the primary containment allowable leakage of 0.5 percent is that the analytical predictions of radiological consequences in the ABWR SSAR is based on the assumption of a containment leakage of 0.5% per day plus a separate MSIV leakage. This approach has been reviewed and found acceptable by the USNRC.

No change is required to the PSAR from the response above.

4. The ABWR SSAR effective flow area from the RPV side (Figure 6.2.2) and feedwater flow from the feedwater side (Figure 6.2.3) includes modeling of flow conditions during the initial inventory depletion periods, as noted on these figures. Blowdown flow during inventory depletion period is controlled by the rate at which the initial fluid inventory in the pipe decompresses and flows through the break, which is generally less than the final steady critical flow. As a simplified, and slightly on a conservative side, modeling approach, these inventory depletion periods which are very short were ignored in the PSAR containment pressure response calculations. However, as observed from the ABWR SSAR and the PSAR results, the peak pressure response results are rather insensitive to the modeling of initial inventory depletion periods.

   No change is required to the PSAR from the response above.

5. No, use of the Mark III containment references does not affect the accuracy of ABWR containment analysis. The analytical models described in References 6.2-1 and 6.2-3 basically simulate the response of a pressure suppression containment to a postulated loss-of coolant accident, and they are equally applicable to all BWR containment analyses including the ABWR containment analysis. These analytical models have been found to be acceptable to USNRC for their applicability to Mark I, Mark II and ABWR containment analyses.

   No change is required to the PSAR from the response above.

6. The following paragraphs provide a discussion of the transfer of lower drywell air content into wetwell during a feedwater line break (FWLB) accident.

   Because of the unique containment configuration, transfer of the lower drywell air will start after the peak drywell pressure occurs. As noted in 6.2.1.1.3.3.1 (16), the drywell is modeled as a single node. As a conservative approach, the model assumption takes credit for transfer of 50% of the lower drywell (LD) air content into the wetwell air space, starting right after clearing of the vent system and simultaneously with transfer of the upper drywell (UD) air. As such, the single node drywell volume represents 100% of the UD volume plus 50% of the LD volume. Accordingly, this model assumption case makes available and allows more air transfer to the wetwell air space during the initial period (where drywell is pressurizing due to mass/energy blowdown from the pipe break and the pressure reaches to its peak value), compared to the actual case which will have available only the UD air for transfer to the wetwell air space during this initial period. Consequently, the model assumption case will involve transfer of more air to the wetwell air space during the initial period than that expected for the actual case. As a result, the model assumption case will result in higher wetwell airspace pressure leading to higher drywell peak pressure.

Furthermore, as it is obvious from the areas under the actual and model assumption curves in Figure 6.2-5, the LD air available for transfer to the wetwell air space appears to be smaller for the model assumption case than that for the actual case. However, as noted above, the model assumption case conservatively allows transfer of the LD air to the wetwell air space starting right after clearing of the vents which is expected to result in higher wetwell airspace pressurization which, in turn, will lead to higher drywell peak pressure, while the LD air transfer in the actual case will start after the drywell peak pressure occurs. The larger LD air transfer for the actual case will occur during the later part of the transient when drywell pressure is stabilizing to a steady value below the initial peak value as apparent in Figure 6.2-6. However, added effect of this larger air flow is expected to not result in steady pressure greater than that determined by the model assumption case. Therefore, it can be inferred that the model assumption case will produce higher drywell peak pressure compared to that expected for the actual case.

No change is required to the PSAR from the response above.

8. The following response is based on Figure 6.2-18 which is more pertinent than Figure 6.2-8 (noted in the question) for comparison with the results in Figure 6.2-17. Figure 6.2-18 shows differential pressure in the wetwell and drywell relative to the reactor building, similar to the results shown in Figure 6.2-17. Figure 6.2-17 shows the case with no containment spray, and depressurization is due to ECCS flooding from the break; Figure 6.2-18 shows the case where both wetwell and drywell sprays operate with no ECCS flooding.

   Yes, the observed trend in Figure 6.2-17 showing a higher wetwell-to-drywell differential pressure than that in Figure 6.2-18 is reasonable. The drywell and wetwell pressure response results shown in Figure 6.2-17 are for the analysis performed to determine and confirm sizing of WDVBS which will limit drywell-to-wetwell negative differential pressure below its design value. The initial condition for the analysis is when the drywell pressure first drops below the wetwell pressure, as stated on Page 6.2-15. From that point in time, drywell depressurization due to cold ECCS water spilling out of the break was considered and evaluated, recognizing that ECCS water flow will result in more severe drywell depressurization than that due to drywell spray actuation as noted on Page 6.2-14.
   The drywell and wetwell pressure response results shown in Figure 6.2-18 are for the analysis performed to check for the PCV-to-Reactor Building negative pressure differential, as noted on Page 6.2-16. The initial condition for this analysis is the same as that for analysis in Figure 6.2-17, and the drywell depressurization is due to drywell spray actuation. Therefore, considering that ECCS water flow causes a more severe drywell depressurization than that due to drywell spray actuation, the observed trend of higher wetwell-to-drywell pressure differential in Figure 6.2-17 compared to that in 6.2-18 before 75 seconds is reasonable.

   No change is required to the PSAR from the response above.

9. 5.9 kPaD is correct. No change is required to the PSAR from this response.

Further Clarification: to ROCAEC Comment:

As noted in our previous response, the wetwell-to-reactor building differential pressure value of 5.9 kPaD shown in Figure 6.2-18 of the Lungmen PSAR is correct, compared to the 0.06 kPaD value shown in Figure 6.2-18 of the ABWR SSAR. The SSAR value of 0.06 kPaD appears to be a typo. The differential pressure value of 0.06 was originally expressed in kg_f/cm^2 units. Somehow, the numerical value of this differential pressure was not changed to correspond to SI units. A unit conversion will show that 0.06 kg_f/cm^2 is equivalent to 5.9 kPaD.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **I-06-002**

PSAR Sections: **Ch 6.2.1**

Question Date: **December 31, 1997**

PSAR Question:

7. In 6.2.1.1.3.3.1.2 (4), RHR heat exchanger is assumed to start 30 minutes after the accident. However, in SSAR, it is on 10 minutes after the accident. Also, in the same section, the maximum service temperature is assumed to be 30°C, while SSAR specifies 35°C. Please explain both differences.

Response:

7. The assumption of 30 minutes start time will provide slightly conservative pool temperature response results, compared to those with 10 minutes start time. The 30 minutes start time assumption was changed in the PSAR from the SSAR to be consistent with the operation action time assumption in 6.3.1.1.1 (3).

   The assumption of 35°C in ABWR SSAR is representative of site-envelope service water temperature conditions in the US, whereas, the PSAR assumption of 30°C service water temperature is representative of maximum sea water temperature expected for the Lungmen site. There are recent indications that Lungmen site-specific maximum sea water temperature may be higher than 30°C. However, keeping in view inherent conservatism in containment analyses, anticipated increase in maximum sea water temperature is expected to not result in calculated pool peak temperature exceeding the design maximum pool temperature of 97.2°C. The updated analyses based on final design maximum sea water temperature will be provided in the FSAR.

   No change is required to the PSAR from the response above.

Further Clarification to ROC-AEC's Comments:

Yes, if the sea water temperature is higher than 30°C, the suppression pool maximum temperature will be expected to be higher than that based on the 30°C sea water temperature. The maximum pool temperature response with a higher sea water temperature, say 35°C, is expected to not exceed the design temperature of 97.2°C. This is explained further in the following paragraphs.

The ABWR SSAR analyses included parametric studies to make an assessment of the impact of different pool initial and ultimate heat sink temperature values on the pool maximum temperature response. For a bounding case, with pool initial temperature of 43.3°C and ultimate heat sink temperature of 37.8°C, the pool maximum temperature value was found to be 96.9°C. In comparison, for the nominal case, with pool initial temperature of 35°C and ultimate heat sink temperature of 35°C, the pool maximum temperature response was found to be 92.5°C. The suppression pool maximum temperature value of 96.9°C reported in Table 6.2-1 of the ABWR SSAR corresponds to the bounding case.

For the Lungmen design, the long-term pool temperature response is based on the pool initial temperature of 35°C and the sea water temperature of 30°C. The calculated pool maximum temperature with these temperature conditions is found to be 88.9°C , which results from a main steam line break. Therefore, the Lungmen design appears to have adequate margin to accommodate sea water temperature higher than 30°C. For example, with sea water temperature of 35°C and pool initial temperature of 35°C, the pool maximum temperature response will be expected to be 91.5°C, which will still be below the design value of 97.2°C.

The suppression pool maximum temperature value of 96.9°C, and also the wetwell gas space temperature value of 98.9°C, in Table 6.2-1 of the Lungmen PSAR are carryover from the Table 6.2-1 of the ABWR SSAR. The Table 6.2-1 in the Lungmen PSAR will be updated to include the calculated pool peak temperature and the wetwell gas space temperature values consistent with the Lungmen PSAR analysis results. The updated Table 6.2-1 will incorporate the following changes:

Wetwell temperature:

•Gas space	change from 98.9°C	to	**110.3°C**
•Suppression pool	change from 96.9°C	to	**88.9°C**

As a result of the response to this comment/question, the Lungmen PSAR will be updated accordingly.

ROCAEC Review Comments :

(1) Maximum service water temperature should be corrected to 35 degrees C.
(2) PSAR Table 6.2-1 and other relevant PSAR sections should be corrected as well based on this new temperature.
(3) GESSAR Table 6.2-1 calculated suppression pool temperature to be 96.9 degrees C and Lungmen PSAR Table 6.2-1 calculated suppression pool temperature to be 91.5 degrees C. Why so much difference ? (Theoretically, both use Maximum Service Water Temperature of 35 degrees C as basis so the difference should not be that great) If GESSAR uses Bounding Value of maximum Service Water as calculation basis then Lungmen PSAR should have the same calculation as well for comparison purposes.
(4) Please explain what the calculated heat load is in Table 6.2-1 of PSAR ? and which reference plant was used.
(5) The clarifications for the above should be completed before the issuance of CP.

Further Clarification:

(1) Agree. The maximum sea water temperature (or maximum service water temperature) input value for the LOCA response analyses will be corrected from 30°C to 35°C.

(2) Agree. Table 6.2-1 of the PSAR and the related text will be modified to reflect the analysis results based on maximum sea

water temperature of 35°C
(see attached mark-up pages, File 6_2ATACH.PDF). Also, other PSAR sections relevant to sea water temperature parameter are being reviewed and the pertinent text/tables/figures will be identified and updated, as appropriate and applicable, to reflect maximum sea water temperature value of 35°C.

(3) The following provides further clarification on the calculated suppression pool temperature value listed in Table 6.2-1 of the ABWR SSAR. It is assumed that the question intends to refer to Table 6.2-1 of the ABWR SSAR and not the GESSAR Table 6.2-1. GESSAR is for BWR 6 and Mark III plant design.

Technically, the design basis loss-of-coolant accident (LOCA) suppression pool temperature response analyses are plant specific, and they are based on the plant specific site ultimate heat sink (UHS) temperature value. For these analyses, a zero percent exceedance value of the plant site specific UHS temperature is used which is determined from and based on site specific UHS temperature historical data. There exist no USNRC requirements which call for additional margin on the plant specific UHS zero percent exceedance temperature value for the suppression pool long-term temperature response analyses which determine the suppression pool maximum temperature value under design basis LOCA condition. However, utilities, at their own discretion, may elect to include some additional margin in the UHS temperature zero percent exceedance value for use in the long-term suppression pool temperature analyses. Again, there are no USNRC requirements for the additional margin, and the magnitude of any additional margin is strictly determined and defined by the utilities for their plant specific application.

Further, as noted in the earlier response, the suppression pool maximum temperature value of 96.9°C reported in Table 6.2-1 of the ABWR SSAR corresponds to the bounding case with pool initial temperature of 43.3°C and ultimate heat sink temperature of 37.8°C. It is appropriate to again point out here that this bounding case should not be interpreted as representative of calculation basis for the pool maximum temperature response based on the bounding value of maximum service water temperature. Rather, this case was analyzed to demonstrate and confirm technical adequacy of the ABWR design in meeting the design requirement of maximum pool temperature value of 97.2°C for potential plant sites which may encounter maximum service water temperature up to 37.8°C. Plant sites with maximum service water temperature less than 37.8°C will have more margin in the calculated value of their design basis pool maximum temperature response.

In view of the above clarification, it should be apparent that the Lungmen PSAR design basis analysis calculations will need to be based on the Lungmen site specific maximum service water (sea water) temperature 35°C - LOCA analysis calculations based on 35°C have been performed, see Response (2) in above. Therefore, Lungmen PSAR LOCA analysis calculations corresponding to the ABWR SSAR bounding case conditions are not warranted. As clarified in above, there exist no USNRC regulatory requirements for the LOCA analysis calculations based on service water temperature above the site specific maximum service water temperature value. However, during the FSAR work, any need to include additional margin in the maximum sea water temperature value will be determined and incorporated, as appropriate, in the FSAR LOCA analysis calculations.

(4) The calculated values of the containment parameters shown in Table 6.2-1 of the Lungmen PSAR are based on Lungmen specific LOCA analyses which analyzed postulated pipe breaks in feedwater and main steam lines. A LOCA results in most severe heat load on the suppression pool which, in turn, results in most severe pool maximum temperature response. A detailed description of the LOCA analyses is contained in Section 6.2.1 of the PSAR, and key assumptions and initial conditions which determine heat load on the suppression pool are described below.

a. The reactor is assumed to be operating at 102% of the rated thermal power, which maximizes the post-accident decay heat.

b. The ANSI/ANS-5.1-1979 decay heat is used.

c. The suppression pool is at its low water level (minimizes the initial pool mass) and its temperature is the operating maximum temperature (maximizes the pool initial stored energy). This combination of pool initial conditions will maximize the pool temperature response.

d. Maximum (zero percent exceedance) service water (sea water) temperature value is used.

e. ECCS pumps motor rated horsepower is continuously added to the suppression pool.

Basically, the Lungmen PSAR analyses are based on the same containment design configuration as that used in the ABWR SSAR analyses. Therefore, in principle, the Lungmen PSAR analyses results should be expected to be the same as those from the ABWR SSAR analysis results, if both analyses use the same containment configuration, the same engineering computer program, the same modeling assumptions, and the same initial input conditions. However, as noted in our earlier response, the ABWR SSAR analysis calculations result in pool maximum temperature response of 92.5°C with pool initial temperature of 35°C and ultimate heat sink temperature of 35°C, whereas, for the same pool initial temperature and ultimate heat sink temperature conditions, the Lungmen PSAR analysis calculations result in pool maximum temperature response of 91.5°C. The cause for this one degree difference in the ABWR SSAR and the Lungmen PSAR analysis calculations is explained as follows:

The Lungmen PSAR analysis calculations are based on the same engineering computer program as that used for the ABWR SSAR analysis calculations. A detailed review of the ABWR SSAR and the Lungmen PSAR analysis calculations input data revealed that the ABWR SSAR and the Lungmen PSAR analysis calculations use the same assumptions and initial input conditions, except the HPCF pump motor horse power input value.

The HPCF pump motor horse power input value in the ABWR

SSAR analysis calculations happens to be for two loops, when instead, this input value should have been for one loop only, consistent with the single failure assumption which implies loss of one out of two HPCF loops. This modeling of the pump motor horse power makes the ABWR SSAR analysis calculation results unduly conservative. Whereas, the HPCF pump motor horsepower input value in the PSAR analysis calculations is for one loop only, which is consistent with the LOCA single failure assumption and introduces no unnecessary conservatism in the calculated pool maximum temperature response.

In conclusion, given that the pump motor heat is continuously added to the suppression pool in the analysis calculations, the one degree difference in the ABWR SSAR analysis calculations and the Lungmen PSAR analysis calculations is attributed to the unduly conservative input of the second HPCF pump motor horse power input in the ABWR analysis calculations.

(5) The above clarifications are timely for the issuance of CP.

The Lungmen NPS PSAR will be updated as mentioned in the above clarification.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-06-003

PSAR Section: 6.2.2

Question Date: December 31, 1997

PSAR Question:

In 6.2.2.3.1, the containment cooling starts 30 minutes after the accident. Does this containment cooling include containment spray? Please explain.

Response:

No. This cooling considers RHR operating in suppression pool cooling (SPC) mode. The design basis pressure/temperature analysis does not require and model drywell/wetwell sprays.

No change is required to the PSAR from the response above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **I-06-004**

PSAR Sections: **Ch 6.6**

Question Date: **December 31, 1997**

PSAR Question:

1. In Section 6.6 (P. 6.6-1) and 6.6.3.1 (P. 6.6-4), please explain which annual version of ASME B&PV Code Section XI (pre-1989, for example 1986. 1983 version of ASME Code Section XI, Subsection IWD has category D-A, D-B, D-C, but 1992 version has only D-A, D-B but no D-C).

2. In Sections 6.6 (P. 6.6-1), 6.6.2.1 (P. 6.6-3) and 6.6.8 (P. 6.6-9), the statements "The detailed program for pre-service and in-service inspection will be developed <u>later</u>", "The detailed in-service inspection program for the RHR heat exchanger will be developed <u>later</u>" and "The code exemption portions of systems are specifically identified in Table <u>(later)</u>." When can you submit ? Please specify the date.

3. In Section 6.6.7.2 (P. 6.6-8) about erosion-corrosion, NUMARC program (in Generic Letter 89-08) and applicable rules of sec. XI of ASME Code were used. Please include the former document (NUMARC) in PSAR appendix, and specifically clarify the related chapter and section of the applicable rules.

Response:

1. Based on the notice of contract-award (NOA) date of June 16, 1996 and in accordance with the applicable January, 1, 1996 Edition of the Code of Federal Regulations 10CFR50.55a, the applicable version of the ASME B&PV Code Section XI, Division 1, is the 1989 Edition. The second sentence of the first paragraph on Page 6.6-1 (Section 6.6) will be modified as follows: "It describes those programs implementing the requirements of ASME B&PV Code Section XI, 1989 Edition, Subsections IWC

and IWD." As this modification implies, for the pre-service examination (PSI) discussed in Section 6.6.3.1 for the ASME Class 1, Class 2 and Class 3 components, ASME Section XI, 1989 Edition, applies. For the edition applicable to the inservice inspection, please refer to the response to Question Track Number N-01-010.

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

2. The identified sentences will be revised to read as follows:
   In Section 6.6 (P. 6.6-1): "The detailed program for pre-service and in-service inspection will be developed based on the final as-built plant configuration, i.e., addressing specific welds, bolting, pipe supports, etc."
   In Section 6.6.2.1 (P. 6.6-3): "The detailed in-service inspection program for the RHR heat exchanger will be developed based on the as-built design, and any inservice inspection program ... application."
   In Section 6.6.8 (P. 6.6.-9): "These portions of systems are specifically identified in Table 6.6-1."

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

3. NUREG-1344, which contains the NUMARC guidelines in its Appendix A, is a public document and, therefore, the guidelines need not be included in the PSAR as an appendix. The PSAR will be revised to list NUREG-1344, Erosion/Corrosion-Induced Pipe Wall Thinning in U.S. Nuclear Power Plants, April 1989, as Reference 6.6-1 in a new Section 6.6.9, Reference. Table 1.8-22 will also be updated to include NUREG-1344. The second sentence in Section 6.6.7.2 will be modified as follows: "The examination schedule and examination methods shall be determined in accordance with the NUMARC program, which is provided as Appendix A in NUREG-1344 (Reference 6.6-1), or another equally effective program, as discussed in Generic Letter 89-08, and applicable rules incorporated in the future in Section XI of the ASME Boiler and Pressure Vessel Code."
   There are no currently applicable rules in ASME Section XI for

inspection of pipe wall thinning. "Applicable rules" currently in Section 6.6.7.2 refers to any rules which might be incorporated into ASME Section XI at some future date and be required as part of an inservice inspection program. The second sentence in Section 6.6.7.2 to be revised as mentioned above clarifies this.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-06-005

PSAR Sections: Ch 1B.2.2.21 & 6.2.5

Question Date: December 31, 1997

PSAR Question:

The 27th line in Page 6.2-69, only 0.72% active clad was considered to experience the metal-water reaction. Does it comply with the NRC requirement in Section 1B.2.21 (1) in Page 1B-8 ? Also, in Figure 6.2.41, does the analysis assumption comply with NRC requirement in 1B.2.21 ? Please explain separately in compliance with 1B.2.21 requirements.

Response:

In the Lungmen NPS, there are no design basis events that result in core uncovery or core heatup sufficient to cause significant metal-water reaction. Therefore, per Regulatory Guide 1.7, which governs the design requirements for the flammability control system, the design basis metal-water reaction is that equivalent to the reaction of the active clad to a depth of 0.0058 mm. This is equivalent to 0.72% of the active fuel for an 8x8 fuel bundle design with a fuel rod outer diameter of 12.27 mm (0.483 inches) and 0.95% of the active fuel for the GE12 10x10 fuel bundle design with a fuel rod outer diameter of 10.26 mm (0.404 inches). The Lungmen NPS is also provided with permanently-installed recombiners which prevent the buildup of oxygen, due to radiolysis, from creating a potentially flammable mixture. PSAR Figure 6.2.41 is based on GE12 fuel bundle design and Regulatory Guide 1.7 assumptions. However, the 0.72% value on PSAR Page 6.2-69 needs to be revised to 0.95%.

The requirements in Paragraphs 1B.2.21 (1) and 1B.2.21 (2) are not applicable to flammability control considerations for the Lungmen NPS primary containment, which is inerted (as discussed in PSAR Section 6.2.5) and is, therefore, protected from hydrogen combustion regardless of the amount or rate of hydrogen generation. In fact, increasing amounts of hydrogen moves the primary containment oxygen concentration further from the flammable regime. The Lungmen NPS is also provided with permanently-installed recombiners which prevent the buildup of oxygen, due to radiolysis, from creating a potentially flammable mixture. Radiolysis is the only potential source of oxygen in the Lungmen NPS primary containment. Paragraph 1B.2.21 (4) is also not applicable as inerting is accomplished prior to the onset of the accident. Regarding Paragraph 1B.2.21 (3) requirements, a detailed discussion is provided in Section 19.4.3.7.1.1 on survivability of equipment necessary to mitigate the consequences of a severe accident involving 100% fuel-clad metal water reaction.

PSAR Page 6.2-69 will be changed to replace "0.72%," in the second paragraph from the bottom, with "0.95%."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-06-006

PSAR Sections: Ch 6.1.2.2

Question Date: December 31, 1997

PSAR Question:

On the 10 RIPs, there is organic material - polyacrylic and polyethlene. Please provide detailed description of the radiolytic and pyrolytic decomposition methods for testing these two materials and the test data.

Response:

The subject materials are used in the RIP motor winding insulation and they are always submerged in water about 50° C. Therefore, pyrolytic decomposition is not a concern. The radiolytic degradation of the electrical properties is a primary consideration. The 20-year life in 6×10^5 gray, 60° C environment is a desired requirement. The RIP vendor will confirm in the future if this requirement is met by the tests chosen by the vendor. The electrical resistance will be checked during the maintenance and inspection of the RIP motor every five years or sooner.

No change is required to the PSAR from the response above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-06-007

PSAR Sections: Ch 6.3

Question Date: December 31, 1997

PSAR Question:

1. Is current U.S. ECCS regulations (10CFR50.46, Appendix K) still applicable to the ABWR LOCA analysis and evaluation modeling ? It is learned that GE still use the conventional LOCA analysis codes (SAFER, LAMB, SCAR). Are they applicable to ABWR ? Please explain.

2. According to the NRC issued SER for the GE Standard SAR, comparison was made between GE long term LOCA analysis code SAFER and GE ABWR full integral simulation test (FIST) facility test data and also with TRACG-P analysis results. Please provide the comparison data.

Response:

1. Yes, the current US ECCS regulation (10CFR50.46 and Appendix K of 10CFR50) is applicable to the ABWR design. The SAFER/GESTR-LOCA licensing methodology, which is contained in GE proprietary report, NEDE-23785-1-PA, October 1984, is applicable to all BWRs including the ABWR, as approved by the USNRC. It allows calculation of the plant-specific break spectrum to be considered using nominal inputs and assumptions. Furthermore, the calculations of the limiting peak cladding temperature (PCT) to demonstrate conformance with the requirements of 10CFR50.46 must include specific inputs and models as specified in Appendix K.

   The BWR LOCA analysis licensing basis PCT is derived by incorporating appropriate margin for specific conservatism as required by Appendix K and a plant variable uncertainty term for the limiting PCT value calculated using nominal input values. To conform with 10CFR50.46 and the SAFER/GESTR-LOCA methodology, the licensing basis PCT for the SAFER/GESTR-

LOCA methodology must be less than 2200°F. Based on the above USNRC approved methodology (NEDE-23785-1-PA) and the fact that the ABWR LOCA analysis results show large margins (no core uncovery) to the licensing limits, the SAFER/GESTR-LOCA analysis evaluation was simplified for the ABWR. In the simplified methodology, break spectrum analysis calculations are performed using Appendix K assumptions (see PSAR Table 6.3-6) to conservatively determine the ECCS performance. The licensing basis PCT value for the ABWR is then determined by using the most limiting case resulting from the Appendix-K required LOCA evaluations.

The key design feature difference between the ABWR and other BWRs is that the ABWR design utilizes internal recirculation pumps. The LOCA analysis licensing models (SAFER and LAMB) that have been used for BWRs have the capability to simulate operation of the internal pump. The TASC code (formerly called SCAT), which models a single BWR fuel bundle, is, therefore, directly applicable to all BWRs including ABWR. Application of these models to the ABWR LOCA licensing analyses has been reviewed and found acceptable by the USNRC under the US ABWR certification program.

No change to the PSAR is required from the above response.

Further Clarification to ROC-AEC's Comments:

This is to clarify that only the methodology for analysis was simplified for the analysis of ABWR LOCA, and there was no simplification of the analytical models in SAFER/GESTR. This is explained in more detail in the following paragraphs.

The SAFER/GESTR-LOCA application methodology is based on the generic studies presented in NEDE-23785-1-PA. The "simplified methodology" for the Lungmen LOCA analysis uses the same LOCA evaluation models as documented in NEDE-23785-1-PA. To reduce the amount of calculations the simplified methodology takes advantage of the low calculated peak cladding temperatures (PCTs) for Lungmen and takes a slightly more conservative approach. The difference in the methodology is in only two areas.

The first difference is in the break spectrum calculation. The methodology described in NEDE-23785-1-PA uses nominal input assumptions to determine the break spectrum results and then recalculates the limiting cases using Appendix K input assumptions. Since the PCTs are so low for Lungmen, the break spectrum was directly calculated using Appendix K input assumptions.

The second difference is in the calculation of the bounding PCT. The methodology described in NEDE-23735-1-PA varies the plant uncertainty parameters one at a time and determines its affect on the limiting case PCT. Then the results are combined in a statistical manner to determine a bounding PCT. Again, since the PCTs are so low for the Lungmen design, the most limiting case was analyzed with all the plant uncertainty parameters set at their conservative values to determine a bounding PCT value.

The Licensing Basis PCT is determined by using the most limiting case resulting from the above Appendix K evaluations. Even with this conservative approach the licensing PCT for Lungmen with GE12 fuel is only about 521°C which is well below the 1200 °C licensing limit.

The GE-NE LOCA analysis licensing models (SAFER and LAMB) to perform SAFER/GESTR-LOCA analysis for all BWRs were utilized for the ABWR LOCA analyses. Since the models have the capability to simulate operation of the internal recirculation pump, which is the key design feature difference between the ABWR and other BWRs.

2. A detailed comparison of SAFER against FIST-ABWR data and TRACG is described and contained in GE proprietary report, NEDE-30996P-A, October 1987. GE has provided a copy of this report to TPC by Letter LOTP-1998-0006, dated January 19, 1998.

No change to the PSAR is required from the above response.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: I-06-008

PSAR Sections: Ch 6.5 Fission Product Removal and Control

Question Date: December 31, 1997

PSAR Question:

1. Please add the definition, scope and requirements of primary and secondary containment. During normal operation, the secondary containment is maintained at 62Pa negative pressure and what is the pressure of the primary containment ? Any regulations or evaluation results for this pressure ? Please explain.
2. Does the periodic test for the charcoal filter include absorption saturation test ? (besides Iodine and noble gases, steam and other gases will be absorbed too which will affect its efficiency)
3. Will heating up of the charcoal filter release some of the absorbed gases ? Please clarify.
4. When evaluating SGT during LOCA, please clarify if simultaneous generation of the maximum value of iodine and noble gases were used for the sizing of the charcoal filter.
5. Is it regulatory requirement that CRHA contain the supplies of food and necessities for 5 people over a 5-day period ? and under emergency conditions, CRHA HVAC system have the capability to sustain continuous habitability of 12 people ? Are there any law requirements or is it based on the appropriate estimates ?
6. We suggest adding rules/guidelines in regards to the periodic inspection and replacement of perishable food items.
7. What is the basis of the assumption or proper evaluation that under accident conditions, 100% noble gases and 1% iodine of the fission products will enter CRHA or MCAE ? Is it also the design basis for the CRHA or MCAE flow rate under accident conditions ?
8. We suggest adding setpoints for radiation & toxic gas detection system in the CRHA or MCAE and setpoints for automatic and manual operating systems should also be included.
9. We suggest adding inleakage rate limits for damper or valves.

Response:

1. As required by Regulatory Guide 1.70, PSAR Section 6.5 addresses the definition, scope and requirements of fission product removal and control systems for primary and secondary containment. However, as requested in the question, the following sentences will be added at the beginning of Section 6.5 to provide an overall introduction. "This section provides information on the fission product removal and control systems for the primary and secondary containment. The filter systems used for removal of the fission products from the primary and secondary containment and the control room habitability area are described in Subsection 6.5.1. Section 6.5.2 discusses fission product removal by containment spray, and Section 6.5.3 addresses fission product control via the primary and secondary containment. For the functional design information (such as the definition, scope and requirements) of the primary and secondary containment, refer to Subsections 6.2.1 and 6.2.3, respectively. The information on other features of the containment systems is provided as follows: Subsection 6.2.2 on Containment Heat Removal System, Subsection 6.2.4 on Containment Isolation System, Subsection 6.2.5 on Combustible Gas Control in containment and Subsection 6.2.6 on Containment Leakage Testing"
   The normal internal PCV pressure is maintained at less than or equal to 5.20 kPaG (0.75 psig), a technical specification requirement (PSAR Section 16.3.6.1.4). The basis for this requirement is discussed in Section 16B.3.6.1.4. As mentioned in PSAR Paragraph 6.2.5.1(5), the primary containment pressure is maintained at a positive pressure, with respect to the secondary containment, to prevent air leakage into the PCV, thus diluting the nitrogen atmosphere.
   A change to the Lungmen NPS PSAR will be made as a result of the response above.

2. In accordance with Section 15 of ASME Standard N510, Testing of Nuclear Air-Cleaning Systems, samples of the charcoal absorber will be periodically withdrawn and tested, in a laboratory

setting, to determine the overall efficiency of the entire bank for the retention of radioiodine. Routine batch testing is required to assure the efficiencies given in Article FF-5200 of ASME AG-1. PSAR Section 6.5.1.4 references ASME N510 for performing tests. No change to the Lungmen NPS PSAR is required as a result of the response above.

3. The SGT system is not expected to adsorb any gaseous fission products. Therefore, there should not be any re-release of these due to an abnormal temperature escalation.
   No change to the Lungmen NPS PSAR is required as a result of the response above.

4. The sizing of the SGT filters is not dictated by the LOCA iodine and noble gas concentrations, but instead, is based on the needed flow rate capacity. The HEPA filters are sized according to MIL-F51068F, based on the flow rate that the SGT system is expected to handle for a worst case LOCA event. The basic parameters used in sizing the SGT filters are discussed in PSAR Section 6.5.1.3.2.
   No change to the Lungmen NPS PSAR is required as a result of the response above

5. There are no specific requirements for the number of people and the quantity of expendable supplies required to maintain the people in CRHA. During the ABWR SSAR preparation, this paragraph originally referred to supplies necessary to maintain the environment for a "prolonged period." The NRC requested that a specific time period be specified, and five days was determined to be an acceptable period to the reviewers at the NRC. As to the number of people required in the control room, two separate documents bear upon the subject: (1) Standard Review Plan 6.4 (NUREG-0800) Section II paragraph 7 requires "Self-contained breathing apparatus for the control room personnel (at least 5 individuals) should be on hand," the implication being that the minimum control room population could be five individuals, and (2) NUREG-0737 provides "guidance" in section 1.A.1.3 "Shift Manning" that for a two unit site with two control rooms, the

minimum personnel should be 1 shift supervisor, 2 licensed senior reactor operators, 3 licensed reactor operators, and 3 auxiliary operators between the two control rooms. Therefore, five persons per room was chosen as a reasonable average for manning two units.
The CRHA HVAC System can maintain habitability for 12 people continuously. If the HVAC System is in outside air isolation mode due to exterior smoke, carbon dioxide buildup may occur which would require outside air to be introduced into the CRHA after approximately 72 hours. See the third paragraph of PSAR Section 6.4.4.2 for greater detail.
No change to the PSAR is required from the above response.

6. Providing procedures for inspection/replacement of emergency supplies would seem to be a prudent and necessary action but is an operational consideration. The appropriate document to contain these guidelines will be evaluated during design. The guidelines will be provided in the FSAR or other appropriate document.
No change to the PSAR is required from the above response.

7. The assumptions used to evaluate the suitability of the control room systems under accident conditions are those given in Chapter 15 for the design basis accidents. Of these design basis accidents, only the large break LOCA (Section 15.6.5) is directly referred to by SRP 6.4 for use in determining the adequacy of the control room systems, because only the LOCA produces a long term release. In the LOCA, the core inventory releases are 100% of the noble gases and 50% of the iodines. The PSAR Paragraph 6.4.2.5.1(3) assumes an unspecified leakage pathway into the reactor building, which is treated by the SGT, and of which 99% of the iodines are removed prior to release from the reactor building. Therefore, for a conservative estimate, 100% of the noble gases and 1% of the iodines transported via the reactor building pathway are assumed to be released to the environment and are considered in the control room analysis. As to the flows, the flow rates are primarily dictated by the positive pressure kept in the control room, and potential leakages, such as due to the personnel requirements to enter and exit the control volume.

No change to the PSAR is required from the above response.

8. The setpoints of radiation detection in CRHA will be specified in the FSAR, Chapter 16, Technical Specifications. The setpoints for toxic gas detection will be developed during detailed design and will be provided in the FSAR. Setpoints for automatic and manual systems will be developed during detailed design and will be provided in the FSAR. The following sentence will be added at the end of PSAR Section 6.4.4.2: "Protection from chlorine gas or other chemical releases will be provided if determined to be necessary by analysis (to be provided in the FSAR). Protection will be in the form of detectors and automatic isolation from outside air and by individual respirators stored in the control room for personal use."
   No change to the PSAR is required from the above response.

9. Permissible leakage rates will be determined during detailed design and will be added to the FSAR. No change to the PSAR is required.
   No change to the PSAR is required from the above response.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 06-009

PSAR Sections: Ch 6.2.2

Question Date: December 8, 1997

PSAR Question:

At Kuosheng NPS, the containment spray system (CTMT Spray) A/B will be automatically activated after 10/11.5 minutes delay under drywell high pressure and containment high pressure conditions (SRP 6.2 has similar rule). However, in ABWR, that system will not be automatically activated. Please explain the reason and clarify if this will increase the chance of Early Failure of containment. (As far as BWR-6 EOP is concerned, maintaining containment integrity is the highest priority which can be seen from the fact that once CTMT Spray is activated in the RHR loop, other operating modes will be excluded which shows the importance of CTMT Spray. Also, in an emergency, many alarms might go off in the control room. Operators will probably be very busy for the first 30 minutes. Human errors will be increased if operators are counted on to lineup and start the CTMT Spray system. Why such change in lineup design ?)

Response:

For the ABWR design, the containment peak pressure during a design basis accident remains below its design pressure with adequate margin, with no credit required for drywell/wetwell spray actuation in the analysis. Accordingly, automatic actuation of sprays is not warranted for the ABWR containment design, and there should be no threat of containment failure during a design basis accident condition. The ABWR design specifies a design pressure value of 310 kPaG (45 psig) for both drywell and the wetwell regions, compared to the containment design pressure of 15 psig for Mark III (Kuosheng) design.

The Standard Review Plan (Appendix A to SRP Section 6.2.1.1.C) requires automatic actuation of wetwell sprays in BWRs to mitigate the consequences of steam bypass of the suppression pool. The

ABWR steam bypass analysis calculations do not suggest need for automatic actuation of wetwell sprays, and, hence, the ABWR design deviates from the SRP requirement of automatic actuation of wetwell sprays for BWRs. This deviation from the SRP requirement was reviewed and found acceptable by the USNRC under the ABWR certification program. This deviation is identified in Table 1.8-6 of Chapter 1 in the Lungmen NPS PSAR, as well as the ABWR SSAR.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 06-010

PSAR Sections: Ch 6.2, 6.3, etc.

Question Date: December 8, 1997

PSAR Question:

In this chapter, kPa has been used for pressure units which is another new unit besides the kg/cm^2 or psi that has been used at other nuclear plants in Taiwan. What is the reason ? Please also explain what pressure unit is going to be used by the meters in the Lungmen control room?

Response:

The use of kPa for pressure units is consistent with the TPC's choice of SI units for the Lungmen project. Accordingly, the meters in the Lungmen control room will read and display pressure measurements in SI units.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 06-011

PSAR Sections: Ch 6.3.2

Question Date: December 8, 1997

PSAR Question:

1. Five (one of them also serves ADS purpose) out of the 16 SRVs at Kuosheng NPS have the Low-Low Set capability/function (pressure setpoints for SRV opening and closure are different which reduced the valve hunting phenomenon due to the hysteresis concept). Please clarify if such function/capability is maintained for Lungmen SRV/ADS ?

2. The Rupture Disk at the steam discharge piping in the original design of RCIC has been removed and replaced by a higher grade of pressure resistant steam discharge piping. Would it affect the RCIC operational reliability as a result ? (Because if Rupture Disk is present, the scram probability due to steam discharge piping high pressure will be less and the RCIC reliability will probably be increased.)

3. Please clarify that when LPCF is activated, if all its flow will go through RHR heat exchanger ? If yes, then it is suggested that evaluation be performed to confirm if positive reactivity will be added during an accident and possibly make reactor critical ? (In BWR-6, LPCI mode will not flow through RHR heat exchanger.)

Response:

1. The SRV/ADS design configuration of the Lungmen NPS indirectly maintains beneficial aspects of low-low set capability/function, though the design does not explicitly employ low-low set logic circuitry similar to that employed in the Kuosheng NPS design. The Lungmen design includes 18 SRVs, and eight of the 18 SRVs are designated for ADS function. They are divided into six relief pressure setpoint groups (1, 1, 4, 4, 4, 4),

and five spring pressure setpoint groups (2, 4, 4, 4, 4). In comparison, Kuosheng NPS design has a total of 16 valves with three relief pressure setpoint groups (1, 8, 7), and three spring pressure setpoint groups (7, 5, 4). This grouping of SRVs into different distinct pressure setpoint groups will reduce the valve hunting phenomenon due to the hysteresis.

2. The replacement of the rupture disk by a higher grade of pressure resistant steam discharge piping will not affect the operational reliability of the RCIC. Originally, the design intent of rupture disks was to protect the low pressure exhaust side of the RCIC turbine case and the exhaust line from overpressurization. The upgrading of the exhaust line and the turbine case design for full reactor pressure prompted removal of rupture disks, and this design upgrade results in elimination of potential break locations on the exhaust side inside the RCIC room. There is no scram due to RCIC turbine high discharge pressure but the RCIC turbine is tripped due to high discharge pressure, consistent with the design of earlier BWRs.

3. Yes, when the low pressure flooder (LPFL) mode of RHR is activated, all of its flow goes through RHR heat exchanger before it flows into the reactor pressure vessel (RPV). The water source for the LPFL flow to the RPV comes from the suppression pool (SP). During an accident condition, the core will be subcritical due to insertion of the control rods. The core is designed to have a 1% or higher shutdown margin, which is evaluated at 20 ℃, no void, Xenon free conditions, and the strongest-worth control rod is assumed to be fully withdrawn. During an accident condition, the injection of cold water into the RPV will add some positive reactivity. However, the temperature of the injected water, which comes from the SP, is higher than 20 ℃ and all the control rods are inserted. The reactor will, therefore, remain subcritical with more than 1% shutdown margin. Hence, no confirmatory analysis is warranted to evaluate the effect of this positive reactivity insertion.

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

ROCAEC Review Comments:

1. Reason for not accepting : The 18 SRV at Lungmen is grouped into 6 Relief Pr. Setpoint Groups and 5 Spring Pr. Setpoint Groups which is more refined than Kuosheng but still can not explain away if the Lowest Setpoint Group SRV can overcome the on/off Hunting phenomenon. Please explain for the Lowest Setpoint Group SRV.

Further Clarifications:

1. The Lungmen SRVs logic is designed to mitigate the postulated thrust load concern of subsequent SRV actuations during a SRV open/close event in response to the change of reactor pressure. The Lungmen SRV design configuration does maintain beneficial aspects of the low-low set capability/function as in Kuoshing per the following design features:

(1). The Lungmen SRVs are arranged with more groups as follows:

A. <u>Spring-Action Safety Function</u>: The SRVs are divided into five setpoint groups. The grouping is 2-4-4-4-4 with two at the lowest setpoint.

B. <u>Power-Actuated Relief Function</u>: The SRVs are divided into six groups. The grouping is 1-1-4-4-4-4 with only one valve assigned to the group with the lowest pressure setpoint. The second group is also assigned with one valve with the next lower pressure setpoint. The remaining valves are divided into four groups with equal numbers of valves.

C. <u>Automatic Depressurization System Function</u>: Eight of the 18

SRVs are designated as ADS valves, and are capable of operating from either ADS LOCA logic or overpressure relief logic signals. The ADS valves are divided in two groups with equal number of valves.

The two lowest set valves will lift due to trip setting, and then the reactor pressure blows down and the valve recloses. The nominal blowdown range for Lungmen is 517 kPa (75 psi), these lowest set valves are located in opposing azimuths in the suppression pool to ensure pool temperature mixing and hydrodynamic force separation.

(2). Because Lungmen SRVs have incorporated a > 15% simmer margin (with respect to main steam normal operating pressure), the SRVs will stay closed at rated reactor pressure. Simmer margin is defined as the difference between reactor normal operational pressure and SRV direct actuation pressure (based on nominal settings).

Because of the above SRV design features that have been included in the system, it is concluded that the lowest set SRVs can overcome the on/off hunting phenomenon.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 06-012

PSAR Sections: Ch 6C.3

Question Date: December 30, 1997

PSAR Question:

1. In Section 6C.3 of GESSAR, the requirements on the sizing of the strainer depend on the quantity of the debris that might plug the strainer and multiplied by 3. However, in the same section in PSAR, the sizing of the strainers is such that they will not become more than 50% plugged following 100 days of post LOCA operation. Which requirement is more conservative ?

2. In Section 6C.3 of PSAR, it was mentioned that the sizing of the strainers ... "plus margin"; does this refer to the "multiplied by 3" or "not become more than 50% plugged following 100 days of post LOCA operation" in the above question or others ? If it is others, what is it ? and what is the assumption involved ?

PSAR Response:

1. The first of the two requirements mentioned in the question is not necessary for the Lungmen NPS. The initial sizing of the RHR strainer, which was adopted for the SSAR, was consistent with RG 1.82, Rev. 1, and applied a multiplication factor of 3 (see the ABWR FSER, NUREG 1503, Section 6.2.1.9). The NRC, subsequently, issued Revision 2 of Reg. Guide 1.82, which allows the strainer blockage to be based upon estimation of the amount of debris and its transportation. The Lungmen NPS applies the new RG 1.82 and in parallel applies a requirement that the strainers will not become more than 50% plugged following 100 days of post LOCA operation, which is consistent with the US utilities requirements for the advanced light water reactor design and the TPC Bid Specification requirement. The PSAR Section 6C.4 will be revised and item (7), "The RHR suction strainers will apply an additional factor of 3 design margins" will be deleted.

2. The margin refers to 10% margin in the NPSH available from the static head of the suppression pool. This is discussed in NUREG 1503, Section 6.2.1.9. The 10% margin in the available NPSH is combined with the margin due to the strainer blockage criteria of the above response to size the strainers.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 06-013

PSAR Sections: Ch 6.2.1

Question Date: January 22, 1998

PSAR Question:

SRP Section 6.2.1.1 specifies that all vacuum valves should be operability tested at monthly intervals but in Section 6.2.1.1.5.6.2 of Lungmen PSAR, it was only committed that vacuum breakers will be tested during each outage (it was explained in PSAR that the Actuator material and design have been improved so that the monthly testing is deemed unnecessary). This change from monthly testing as described in SRP to testing during outage is quite a big change in test intervals. Has any PRA or similar analysis been performed as a basis for this change ?

PSAR Responses:

Operating experience with vacuum breaker valves indicates that the air operator used in monthly stroke testing is one of the root causes of failure to close these valves. The experience also indicates that most of the failures of vacuum breaker valves similar to the ABWR design are failure to pass leak-rate tests during refueling and maintenance outages, which would not be detected by monthly stroke testing. Based on this experience and recognizing that the ABWR design retains provision for instrumentation necessary to provide the plant operators with continuous surveillance of the vacuum breaker position, the USNRC and GE agreed to delete the monthly operability testing. This is consistent with the ABWR SSAR (Section 6.2.1.1.5.6.3) and the ABWR FSER (NUREG-1503, Section 6.2.1.8.5).

An actual quantitative estimate of the improved reliability without monthly operability testing has not been determined, since there is no vacuum breaker operability experience without monthly testing. However, elimination of monthly stroke testing, combined with improvements in the valve materials and hinge design, is expected to improve operational reliability of vacuum breaker valves.

ROCAEC Review Comments:

Reason for not accepting : The response was not to the question. Please clarify whether the Actuator of the Vacuum BRK can only be tested during outage but not during normal operation (CTMT filled with nitrogen so personnel entry is excluded). So the Test Frequency can not be changed from what the SRV stipulated as Monthly to Seasonally or Semiannually but should be changed directly to Tested Each Outage ?

Further Clarifications:

This is to clarify that vacuum breaker actuator (i.e., air actuated cylinder) required for remote operability tests will not be included in the Lungmen vacuum breaker system design, consistent with the US ABWR SSAR.

Therefore, given that the Lungmen design will not have air actuated cylinders, the operability tests can only be performed manually during plant outage, but not during plant normal operation. Personnel entry during plant normal operation is excluded because of inerted containment. Accordingly, the functional test of each vacuum breaker will be performed every outage, as stated in the Lungmen Tech. Spec (Chapter 16, SR 3.6.1.6.2).

No change in the Lungmen NPS PSAR will be made as a result of this clarification.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 06-014

PSAR Sections: Ch 6.2.4

Question Date: January 22, 1998

PSAR Question:

SRP Section 6.2.4 specifies that one isolation valve each should be on the inside and outside of the containment (CTMT) penetration piping. But the ACS (Atmospheric Control System) of Lungmen has both these CTMT isolation valves on the outside of the CTMT (PSAR Section 6.2.4.3 stressed the accessibility consideration). This, not only it breaches the SRP rule, but will possibly result in CTMT Bypass condition if cracking occurs upstream of the first isolation valve outside the CTMT. Please explain the nuclear safety impacts with this design.

Response:

Please refer to PSAR Sections 6.2.4.3.2.2.2.3 through 6.2.4.4 These PSAR Sections discuss the above question in detail and the information provided has been reviewed and approved by the US Nuclear Regulatory Commission. The arrangement in question has been standard practice for earlier GE containment structures such as the Mark I, II, and III.

The primary reason these valves are located outside containment is that space inside primary containment is limited, and a significant space allocation is required for placement of, and maintenance of, these large valve/actuator assemblies.

The piping from containment, to and including the valves, is an extension of the primary containment boundary and is designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Class 2 requirements. Since this piping is considered an extension of the primary containment boundary, it is no more likely to fail than any other part of the primary containment boundary. The arrangement of the isolation valves and connecting piping is such that a single active failure of an inboard valve, or a single active or passive failure in the connecting piping or an outboard valve, cannot prevent isolation of the ACS containment penetrations.

We agree that SRP Section 6.2.4 indicates that one isolation valve each should be on the inside and one on the outside of the containment penetration, however this reference is based on the requirements of Title 10, United States Code of Federal Regulations, Part 50, Appendix A, General Design Criterion 56. Please note that 10CFR50, Appendix A, GDC 56 states "each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves....*unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other design basis.*"

The design of the ACS containment isolation satisfies these criteria. No changes to the PSAR will be made as a result of the response to the comment/question.

ROCAEC Review Comments:

Reason for not accepting :

1. Please clarify whether the CTMT Iso. Vlv's of the ACS at K-6/K-7 in Japan are all on the outside of the CTMT ?

2. Nitrogen addition system at Chinshan seems to be on the outside of CTMT (CTMT Iso. Vlv's) too. Please verify !

3. NMP-2 Nitrogen addition system seems to have one either side of the CTMT. Any operational difficulties ? Why Lungmen does not follow it ? Any difficulties ?

4. Please clarify whether CTMT Iso. Vlv's of ACS can all follow Remote - Manual Actuation ? (required by NUREF-1503 FSER/NRC)

Note : RHR P'p and Min. Flow & Test Lines of HPCF P, RCIC T Exhaust & P'p Min. Flow Lines, CTMT Iso. Vlv's of SPCU Suction & Exhaust Lines can all be Remote-Manual Actuated. Is Lungmen the same ? Is ACS the same ?

5. The same question was one of the open items in NRC Review of GESSAR. From SER and GESSAR Ch. 20, it seems GE has provided enough clarifications so NRC closed this issue. Please provide the clarification information GE submitted to NRC (including correspondences).

6. Does CTMT Failure cover 50 cm large pipe break (since the largest pipe in ACS is 50 cm diameter, and the Iso. vlv's are the same as CTMT as class 2) ? If yes, please show where it can be found in PSAR.

Further Clarifications:

1. Yes, they are all on the outside of the CTMT at K6/K7.

2. Yes, Nitrogen addition system at Chinshan has both isolation valves on the outside of containment

3. There has been no difficulties reported so far in the NMP-2 case. The Nitrogen addition system has isolation valves on either side of the containment. However, the Lungmen NPS design (like the rigorously reviewed K6/K7 design and practices in many other plants, such as Chinshan) has both isolation valves outside of the containment, because this arrangement provides improved accessibility for inspection and testing during reactor operation and subjects the valves to mild environment for improved performance. As mentioned in the initial response, the primary reason these valves are located outside containment is that space inside primary containment is limited, and a significant space allocation is required for placement of, and maintenance of, these large valve/actuator assemblies. As noted in clarification to Item 5 below, the NRC wrote in Draft Final Safety Evaluation Report

(DFSER), "The staff believes that this arrangement will improve reactor operational safety."

Furthermore, the ABWR design includes an additional feature in the ACS system of containment overpressure protection system (COPS) for severe accident mitigation as discussed in PSAR Section 6.2.5.2.6.2. A pressure relief line, which should be communicating with the wetwell air space, is connected to the purge exhaust line between the containment and the normally closed first (so called inboard) isolation valve. A special penetration would be required for the COPS relief line if the first isolation valve were to be inside the containment. The Lungmen design allows to eliminate this penetration.

The Lungmen design meets the USNRC requirements that allow precluding a breach of piping integrity and consider single-failure criterion and protection against the effects of flooding, missiles, pipe whip and jet impingement. Thus, as discussed and accepted by the USNRC in the FSER, NUREG-1503, Pages 6-26 and 6-27, the Lungmen design, like the ABWR design, meets GDC 56 on another defined basis as allowed by GDC 56 (see words in italics in the original response).

The clarification below in item 6 addresses the requirements that are satisfied in order to preclude breach of piping integrity.

4. The Lungmen NPS isolation valves, like the ABWR valves, in RHR and HPCF minimum flow and test lines, RCIC turbine exhaust and minimum flow line, and SPCU suction and discharge line are capable of remote-manual actuation in the control room. Similarly, the ABWR as well as Lungmen ACS valves are also capable of remote-manual actuation from the control room. The ACS containment isolation valves will be automatically closed after the containment isolation signal has been received. The remote-manual actuation capability for the above valves has not been changed for Lungmen.

5. The interactions between GE and the NRC were as follows on the

issue of meeting GDC 56 on another defined basis (Initial NRC Question 430.42, DSER Open Item 18, DFSER Open Item 6.2.4.1-1 and close out of the item in FSER, NUREG-1503).

- **NRC Question 430.42 on Isolation Valve Arrangement**

The NRC Question 430.42 (SSAR Page 20.3.2-35) asked for description of isolation valve arrangement for the ACS purge supply and exhaust lines and discuss design conformance with Branch Technical Position CSB 6-4, "Containment Purge During Normal Operations.". By a letter from Dino C. Scaletti (NRC) to J.S. Gay (GE) "Request for Additional Information Regarding the General Electric Company Application for Certificate of the ABWR Design, dated July 7, 1988, the NRC provided Question 430.42.

- **GE Response to Question 430.42**

During the NRC's early review period until about October 1991, GE answered the questions and submitted the SSAR amendments as necessary by numerous correspondences. The practice was to record the responses in SSAR Chapter 20 (the answer to Question 430.42 being located with the question on SSAR Page 20.3.2-35). The GE's response to the NRC Question 430.42, and the NRC Question 430.42.are presented in Attachment 1.

GE described the arrangement of the ACS purge supply and exhaust lines and responded that the purge supply line has two isolation valves in parallel, located outside of, but as close as possible to the primary containment. The response stated further that the exhaust line has a similar parallel arrangement for the two valves located nearest to the containment.

- **NRC Questions on Compliance with GDC 56 (Open Item 18) in DSER**

The NRC reviewed the GE response and identified an open item in an NRC's draft safety evaluation report (DSER). This report was provided with the NRC letter SECY-91-355 (October 31, 1991), and this open issue was identified as Open Item 18. The NRC wrote:

"the isolation valve configuration does not comply with GDC 56, which requires one isolation valve inside and one isolation valve outside containment for each penetration."

- **GE Response to Open Item 18**

GE addressed the Open Item 18 by responding to the NRC in Letter MFN No. 060-92, dated March 11, 1992. –The GE response was as follows:

"Purge and vent valves are currently licensed with both inboard and outboard isolation valves located outside primary containment so they are not exposed to the harsh environments of the wetwell and drywell and are accessible for inspection and testing during reactor operation."

- **NRC Accepts GE Explanation, Requests for more Justification in DFSER, Reclassifies Open Item 18 as Open Item 6.2.4.1-1**

In a subsequent document, NUREG-1469, "Draft Final Safety Evaluation Report," (DFSER), dated October 1992 and issued with NRC Letter SECY-92-349, dated October 14, 1992, the NRC identified the previous Open Item 18 as Open Item 6.2.4.1-1, and wrote as follows:

"GE proposes to design the ABWR purge and vent valves with both isolation valves located outside the primary containment to avoid the harsh environment of wetwell and drywell and to remain accessible for inspection and testing during reactor operation. The staff believes that this arrangement will improve reactor operational safety. However, GE should still justify how the isolation barriers design can protect the containment integrity. GE has agreed to provide the design

information in a future SSAR amendment. This is Open Item 6.2.4.1-1."

• **NRC Closes Open Item 6.2.4.1-1 in FSER**

GE responded to the NRC request for the justification as noted in the DFSER by providing the information in SSAR, Section 6.2.4.3 (Amendment 34). The NRC reviewed and accepted GE's justification, and wrote as follows in the FSER, NUREG 1503:

"The staff finds that the containment isolation provisions for the purge valves are in conformance with BTP CSB 6-4. However, this valve configuration does not comply with GDC 56, which requires one isolation valve inside and one isolation valve outside the containment for each penetration. This was identified as Open Item 6.2.4.1-1 in the DFSER and as Open Item 18 in the DSER (SECY-91-355). GE stated that purge and vent lines do not extend into the containment and have both inboard and outboard containment isolation valves (CIVs) located outside the primary containment so that they are not exposed to the harsh environment of the wetwell and drywell and are accessible for inspection and testing during reactor operation. SSAR Section 6.2.4.3 (Amendment 34) states, in part, that the CIVs for the atmospheric control system located outside the containment will be located as close to the containment as practical. The piping from the containment up to and including both valves is an extension of the primary containment boundary and is designed in accordance with ASME Code, Section III, Class 2 requirements. The CIVs are protected from the effects of flood and dynamic effects of pipe breaks in accordance with SSAR Sections 3.4 and 3.6. The arrangement of the isolation valves and connecting piping is such that a single failure of an inboard valve, or a single active or passive failure in the connecting piping or an outboard valve, cannot prevent isolation of the ACS. The valves are air operated with a pilot

dc solenoid valve that will fail closed on loss of air or loss of electrical power. The power for the dc solenoids will be supplied from independent electrical divisions.

The staff's position on this issue is that locating both isolation valves outside the containment is acceptable if piping and valve design criteria are sufficiently conservative to preclude a breach of integrity. In general, the isolation barriers should be designed to ESF criteria and protected against floods, missiles, pipe whip, and jet impingement. GDC 56 permits containment isolation provisions for lines penetrating the primary containment boundary that differ from GDC 56, provided the basis for acceptability is defined. The staff concludes that this valve arrangement precludes a breach of piping integrity, meets the single-failure criterion, and is protected against the effects of flooding, missiles, pipe whip, and jet impingement and meets GDC 56 on another defined basis, and is acceptable. Therefore, DFSER Open Item 6.2.4.1-1 is resolved."

Thus, Open Item 6.2.4.1-1 was resolved based upon the GE's justification in SSAR Section 6.2.4.3.

6. The portions of the ACS supply/purge line (50 cm) piping as well as the make-up line (5 cm) piping that are attached to the containment and extend to the furthest isolation valve outside the containment will be designed to meet the requirements in PSAR 3.6.2.1.4.2 for the ASME Code Section III, Class 2 piping. Therefore, no pipe breaks are required to be postulated for these portions of the piping. Please refer to PSAR Sections 6.2.4.3.2.2.2.3 (Page 6.2-56) and 6.2.4.1.1(7) and Table 3.2-1b (System T31, Item 4, Page 3.2-43).

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 06-015

PSAR Sections: **6.2.6 Containment leakage Testing**

Question Date: **December 2, 1997**

PSAR Question:

1. Will the containment leakage test program described in Section 6.2.6 be applicable to secondary containment? If not, what is your strategy regarding leakage test as far as double containment design is concerned?

2. Please provide the value of La.

3. How to justify that + 25% is an acceptable number for comparison between test data and integrated leak test data?

Response:

1. The containment leakage test program described in Section 6.2.5 of Lungmen PSAR applies to primary containment and systems and components which penetrate containment. This is in agreement with 10CFR50, Appendix J which defines primary reactor containment leakage testing for water-cooled power reactors. The functions of primary containment and secondary containment are somewhat different. The primary containment acts as a leakage barrier, after the reactor pressure boundary, to control the release of radioactive material from the fuel in the reactor core under Design Basis Accident (DBA) conditions. On the other hand, the secondary containment encloses some of the primary containment systems, and its function is to control primary containment leakage.

   Per Table 6.2-2d of Lungmen PSAR, the design leakage rate for the secondary containment at DBA conditions is 50%/day of secondary containment free volume. The strategy regarding containment leakage test is to maintain the secondary containment under a slight vacuum. The integrity and operability of the secondary containment is demonstrated by Standby Gas Treatment System (SGT) surveillance test per Subsection 16.3.6.4.1, Secondary Containment of Lungmen PSAR. if the required secondary containment vacuum can be established within specified time and flow with SGT.

2. As stated in Subsection 6.2.6.1.1.1 of Lungmen PSAR, La, the allowable leakage rate at DBA conditions, is 0.5 % by weight of the contained atmosphere in 24 hours. The weight of the contained atmosphere can be calculated using the following formula:

   $W = (P - P_v) / RT$

   Where

   W = Calculated containment dry air mass
   P = Total absolute pressure in the containment
   Pv = Containment atmosphere volume weighted vapor pressure
   R = Constant for dry air (8.3144 joules/gmK)
   T = Absolute containment air temperature

   The value of La in the terms other than the 0.5% by weight of the contained atmosphere in 24 hours as stated above is not provided in the PSAR. This is because the value above is a design limit which is applicable to all containments and systems and components which penetrate containments regardless the different parameters in design, and MSIV leakage rate which should be excluded is not available until testing is performed. The value of La, most of the times in the term of volumetric flow, for use at the site, will be calculated and provided in the procedure for Integrated Leakage Rate Test (ILRT).

3. Based on Appendix C of ANSI N45.4, Subsection 6.2.6.1.1.3, Supplement Verification Test of Lungmen PSAR will be revised to read as follows:

"The accuracy of the leakage rate tests is verified by using a supplemental method of leakage measurement. Verification is obtained by superimposing a controlled and measurable leak on the normal containment leakage rate or other methods of demonstrated equivalency right after the leakage rate test is completed. The difference between the composite leakage and the superimposed known leakage results in a calculated leakage rate. If the result of the leakage measurements obtained prior to the introduction of superimposed leakage in reasonable agreement with the calculated leakage, the accuracy of the containment leakage measuring system is verified and the leakage results validated. Complete descriptive details are found in Appendix C of ANSI N45.4."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 06-016

PSAR Sections: 6.D

Question Date: February 10, 1998

PSAR Question:

It was pointed out in Page 6.2-63 item (5) that the Primary containment isolation valves of the ACS will automatically close on drywell high pressure or reactor low water level 3 setpoint. But in section 6D.1 of page 6D-1, it was stated that the valves will close on high drywell pressure (13.83 kpa) which is inconsistent with the previous statement. Please explain.

Response:

PSAR Section 6.2.5.2.1, page 6.2-63, under the heading "The following modes of ACS operation are provided:" Item 5 states:

> "ACS....primary containment isolation valves....are automatically closed if the drywell pressure is high, or reactor water level 3 setpoint is reached...."

PSAR Section 6D.1, page 6D-1 "Bypass Mechanism through ACS Interconnection" referring to the ACS primary containment isolation valves, states:

> "However, this additional bypass leakage path will close in a few seconds because of automatic closure of these valves upon receipt of a loss-of-coolant-accident (LOCA) signal. These isolation valves are designed to close automatically within 15 seconds after receiving a high drywell pressure (13.83 kPa) signal."

We do not believe there is any discrepancy or inconsistency between these two sections.

Section 6.2.5.2.1 identifies the various isolation signals which will lead to closure of the ACS primary containment penetration isolation valves. As stated in the text, and reported in Table 6.2-7 (page 6.2-130), the valves will close on high drywell pressure, reactor low water level 3 setpoint, or high radiation.

Appendix 6D addresses the particular issue of potential steam bypass leakage through primary containment penetration isolation valves 0022 and 0023 in the event of a pipe break inside the drywell. Steam bypass leakage (from the drywell to the wetwell gas space) is associated only with a LOCA inside the drywell. In the event of a LOCA inside the drywell, the drywell pressure will reach the isolation signal level. Appendix 6D analysis assumes, and takes credit for, closure of these isolation valves on a high drywell pressure signal.

Since the focus of Appendix 6D is limited to pipe breaks inside the drywell, and does not consider other types of failures, other modes of containment isolation such as low reactor water level are not mentioned. Although low water level will also occur during the event, the high drywell pressure signal will occur first and is the basis for the Appendix 6D discussion.

Note that low reactor water level, and not drywell high pressure, is associated with pipe breaks outside the primary containment. As discussed in PSAR Section 7.3.1.1.2 item (3) (iii), "Level 3 is set high enough to indicate inadequate vessel water makeup possibly indicative of a breach in the RCPB or process piping containing reactor coolant, ..." For isolation purposes, the level 3 setpoint is there for a LOCA.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 06-017

PSAR Sections: 6.7

Question Date: March 18, 1998

PSAR Question:

PSAR 6.7 did not list all the design parameters of the nitrogen inerting of the containment (such as system capacity, nitrogen pressure and time to fill to pressure, etc.). Please provide such information. Also, please examine when the containment volume is larger for Lungmen (compared with Chinshan NPS), is there any special difficulties for the nitrogen inerting process?

Response:

Please be advised that Section 6.7 of the Lungmen PSAR discusses the Nitrogen System (N2) which is not associated with containment inerting. Operation of N2 is for the purpose of supplying compressed nitrogen to the relief and automatic depressurization functions of the main steam safety relief valves, and for pneumatically operated devices located within primary containment. N2 does not provide the containment inerting function.

Containment inerting is part of the Atmospheric Control System, (ACS). Please refer to PSAR Section 6.2.5 entitled "Combustible Gas Control in Containment" for a detailed discussion of ACS and containment inerting.

PSAR Section 6.2.5.2.2 entitled "Inerting Equipment," states that ACS will be *capable* of reducing the wetwell and drywell oxygen concentrations from atmospheric conditions to less than 3.5% in less than four hours. Technical Specifications (Chapter 16 LCO 3.6.3.2) requires the wetwell and drywell atmospheric oxygen concentrations to be reduced from atmospheric conditions to less than 3.5% in less than 24 hours after thermal power has reached 15% of plant rated thermal power.

PSAR Section 6.2.5.2.5 entitled "Pressure Control" states that the containment atmosphere will normally be maintained between 5.2 and 8.6 kPa gauge pressure.

PSAR Table 6.2-2 entitled "Containment Design Parameters" indicates the drywell and wetwell net free volume in cubic meters as 7350 and 5960, respectively.

The Lungmen ACS will be designed to preclude any operational difficulties for the nitrogen inerting process.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 06-018

PSAR Sections: 6.2.5

Question Date: March 6, 1998

PSAR Question:

GE ABWR SSAR Section 6.2.5.2.6.1 (7) has specified the Piping materials, design pressures and temperatures, etc. but these information are not available in the Lungmen PSAR. Please explain.

Response:

Adoption of the identified SSAR information for the Lungmen NPS has been under consideration.

The PSAR Section 6.2.5.2.6.1 will be revised to include Item (7) as follows:

"(7) The piping material and its design pressure and temperature will be provided in the FSAR."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 06-019

PSAR Sections: Ch. 6 Sec. 1

Question Date: May 2, 1998

PSAR Question:

1. It was stated in SRP 6.1.1.II.A.1.a.1 on Austenitic Stainless Steel specification that the maximum yield strength of cold-worked Austenitic stainless steel is 90,000 psi but such information is not provided in this section of PSAR. Please clarify.

2. It was stated in SRP 6.1.1.II.A.1.a.3 on Austenitic Stainless Steel specification that the material selection and manufacturing of BWR pressure boundary pipings should follow Branch Technical Position MTEB 5-7 but such information is not provided in this section of PSAR. Please clarify.

3. It was stated in SRP 6.1.1.II.A.1.b.2 on Ferrite Welding specification that the moisture control of the low hydrogen welding materials should follow ASME Sec. III App. D Article NB, NC, ND-2000 and 4000 and AWS D1.1 but such information is not provided in this section of PSAR. Please clarify.

Response:

1. As stated in PSAR Paragraph 6.1.1.1.3.3, GE uses solution annealed austenitic stainless steel. Since cold worked material is not allowed, the yield strength limit for cold worked material was not stated. Hardness and/or strain is controlled during fabrication such that material properties, including yield strength, remain in the range of annealed material. Please also see the response to Part 1 of Track No. 04-025.

2. Please note that the current version of SRP 5.2.3 (MTEB 5-7) mentions that MTEB 5-7 has been superseded by NUREG-0313. Compliance with NUREG-0313, Rev. 2 is discussed in PSAR Section 5.2.3.4.1.1, which is referenced in Section 6.1.1.1.3.1.

3. Since Regulatory Guide 1.70, Revision 3, does not identify a discussion on ferritic steel welding, it is not included in PSAR Section 6.1.1. However, Section 5.2.3.3 includes a full discussion on fabrication and processing of ferritic materials. Also, Section 5.2.1.1 and Table 3.2-3 identify applicable subsections of the ASME III Code.

   It is considered not necessary to restate detailed requirements of the ASME Code in the PSAR. By application of the ASME Code, all requirements contained therein are invoked. Purchase specifications for Lungmen materials and fabrication meet or exceed ASME requirements and PSAR Section 5.2.3.3.4 for moisture control in welding materials.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 06-020

PSAR Sections: 6.1 and 9.1

Question Date: May 4, 1998

PSAR Question:

SRP 6.1.1.II.B.1.b puts the ECCS water quality at : PH = 5.3 ~ 8.6 ; GESSAR 9.1.3 also puts the Fuel pool water quality at PH = 5.6 ~ 8.6. Please explain why in Lungmen PSAR the water quality for Suppression pool and Fuel pool is relaxed to PH = 4.5 ~ 10.

Response:

Section 6.1.1.2 of the Lungmen PSAR requires SPCU water to be maintained at a pH range of 5.3 to 8.6. This matches the SRP 6.1.1.II.B.1.b requirements for ECCS water quality.

For consistency with the above SPCU pH range and with SSAR, Section 9.1.3, the same range of 5.3 to 8.6 will be shown in PSAR Sections 9.1.3.1.2 and 9.1.3.2.2.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 06-021

PSAR Sections: 6.1.1

Question Date: May 4, 1998

PSAR Question:

There are two methods of production of high tensile reinforcing bars steel in Taiwan right now :

(1) Conventional method of adding alloys : the reinforcing bars produced will be more uniform but the cost is higher.

(2) Process controlled reinforcing bars (also called water-tempered reinforcing bars) : common steel nugget undergoing fine tuned process to produce different grain structures at the surface and inside the steel.

Reinforcing bars rods produced by both methods above can have the same strength but their tensile strength and yield strength can be different and also the strength is influenced by the connecting methods. Which type of high tensile reinforcing bar is used in Lungmen concrete containment ? Is water-tempered reinforcing bar used in such application ? Please clarify.

Response:

The Reinforcing Steel specified for Lungmen NPS Units 1 & 2 is the deformed bar meeting the requirements of ASTM A615 "Standard Specification for Deformed and Plain Billet-steel Bars for Concrete Reinforcement". The specified yield strength of the material is 60 ksi (400 Mpa) minimum. Either production method is acceptable as long as the requirements of ASTM A615 are met.

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

ROCAEC Review Comments (Professor S. K. Wu of Taiwan University):

Recent research on water-tempered bars (M.S. Thesis by C.Y. Wang, College of Material Science, Taiwan University, June 1998) showed that it's not difficult to meet ASTM A615 specifications for the water-tempered bars. In other words, ASTM A615 would not be able to distinguish between conventional high tensile strength bars and water-tempered bars. Some of the major differences between conventional high tensile strength and water-tempered bars from C.Y. Wang's thesis are listed below :

(1) Question on ratio of tensile strength and yield strength : both CNS560 and ASTM A706 specified that this ratio must be greater than 1.25. For conventional high tensile strength bars this ratio is around 1.50 and water-tempered bars are around 1.25 - sometimes greater and sometimes smaller than 1.25. ASTM A615 did not specify this ratio, but it is obvious from ASTM A706 that water-tempered bars are only marginally acceptable. Please explain if this will impact the structural design of the civil construction ?

(2) Question on connections of bars : mechanical connectors or gas compressed methods can be used for conventional high tensile strength bars for connections but it is found that while mechanical connectors used for water-tempered bars can maintain certain level of strength but gas compressed method will result in 10% of the connections losing its strength over time which should be noted for bars connections.

(3) Question on bars quality : C.Y. Wang's research found that water-tempered bars produced by two different suppliers have drastically different mechanical properties which is rare among conventional high tensile strength bars. Our survey showed that water-tempered bars are not used widely in the States and Japan. Could it be because of its non-uniform quality problem ? GE is requested to express its views on the uniform quality of water-tempered bars (including why the States does not use water-tempered bars very often).

Further Clarifications:

(1) Water tempered reinforcing steel should not be used in the

construction of any Seismic Category I structure because they lack the ductility and reserve strength of conventional high strength reinforcing steel. If water tempered reinforcing steel was used, the reserve strength counted on in an earthquake, in the PRA or in the Containment Ultimate strength evaluation would not be present. A statement will be added to PSAR Subsections 3.8.4.5.1.2, 3.8.4.5.2, and 3.8.4.5.6 to preclude the use of water tempered reinforcing steel and require the construction to be performed using conventional high strength reinforcing steel.

(2) In the construction specification for safety-Related Building, all bar to bar splices of reinforcing steel are required to be in accordance to the provisions of the ASME B&PV Code Section III Division 2 Subsections CC-2310 and CC-4330. The only splices allowed by the ASME Code and the construction specification are lap splices, arc-welded joints, and the following mechanical splices: taper threaded splices, swaged splices, threaded splices in thread deformed reinforcing rebars. Gas compressed methods are not an ASME approved method for bar to bar splicing of reinforcing steel. The mechanical splicing system selected by the constructor must pass all of the requirements of ASME B&PV Code Section III Division 2 Subsection CC-4333.

(3) Water tempered bars are generally not used in the US and Japan. The main reasons are:

1. Because the outer high strength outer shell is brittle they have a tendency to fail the bend and elongation tests. Their high failure rate increases the fabricators costs over conventional high strength reinforcing steel.

2. Because the bar has a high strength outer shell and a milder lower yield inner core the yield strength of the bar is not clear.

3. Because the bar has a high strength outer shell and a mild strength inner core, the use of threaded mechanical connector makes the splice weaker than the bar. This is a violation of ACI 318, ACI 349, and the ASME code which require the

splice to achieve 125% of the specified minimum yield strength of the reinforcing steel.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 06-022

PSAR Sections: 6.2.3

Question Date: May 4, 1998

PSAR Question:

1. Section 6.2.3.2 of page 6.2-41 stated that the Secondary Containment should maintain a pressure of -62 Pa but this number is different from the -63 Pa described in page 9.4-15 (section 9.4.5.1.1.2). Please clarify.

2. Section 6.2.3.3.1.2.3 (Main Steam Tunnel) stated that the blowout panels which are designed to Seismic I requirements, will allow the double-ended break energy to be vented. Please clarify if this is vented to the atmosphere.

Response:

1. This minor difference between the two numbers in Sections 6.2.3.2 and 9.4.5.1.1.2 can be attributed to inconsistency in unit conversion round off. Originally, the negative differential pressure to be maintained in the secondary containment is expressed in English Units as 0.25 inches (water), as stated in the USNRC SRP 6.2.3. This negative pressure value in SI units comes out to be 62.21 Pa, which is rounded off as -62 Pa. The negative pressure value in Section 9.4.5.1.1.2 will be changed to -62 Pa for consistency with Section 6.2.3.

2. The double-ended break energy is vented from Main Steam Tunnel to the atmosphere via Turbine Building.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 06-023

PSAR Sections: 6.2.4

Question Date: May 4, 1998

PSAR Question:

1. Section 6.2.4.3.2.2.2.3 (Atmospheric Control System) stated that all the SC-2 isolation valves are installed outside the containment in order to facilitate the operators to close the valves. But then it was stated that the arrangement of the valves and connecting pipings is such that a single active failure of an inboard or outboard valve cannot prevent the system from functioning. This two statements seem contradictory (since there are only out board isolation valves in ACS, how can there be single active failure of in board valve ?). Please clarify.

2. Page 6.2-57 stated that the valve closure time of the ACS isolation valve is $\leq$ 20 seconds so it's more stable and therefore does not have to conform to the SRP 6.2.4 requirements of 5 seconds. Please provide evidence that supports this claim.

3. Section 6.2.4.3.6 (Containment Purge System) stated that the isolations valves are all installed outside the containment and common mode failures can be easily avoided because of the redundancy and Criterion 54 does not have to be conformed. Please clarify.

4. Table 6.2-2d did not list if there are two strings of filters in the Secondary Containment but instead listed there are two fans. Please clarify if this is a typographical error.

Response:

1. To clarify, both containment isolation valves are outside primary containment. In the case of ACS, the term "*inboard*" refers to the containment isolation valve that is located closest to the

containment penetration (although still outside primary containment.) The term "*outboard*" refers to the containment isolation valve that is located furthest from the containment penetration.

However, for clarity, the PSAR will be revised as follows: The words "inboard" will be replaced with "inboard (or outside but closest to the containment)" and "outboard" with "outboard (or second closest to the containment)."

2. The PSAR Section 6.2.4.3.2.2.2.3 explains how the ≤20 seconds closure time of the containment isolation is sufficient to meet the intent of SRP 6.2.4 and provide appropriate safeguards to prevent release of contaminants to the environment.

   The following points are to be noted:

   - SRP 6.2.4 states that the 5-second closure speed *may* be necessary to assure that the purge and vent valves would close before the onset of fuel failures following a LOCA.

   - The ACS purge and vent valves are closed during normal plant operation and are open *only* during the inerting (startup) and de-inerting (shutdown) processes.

   - The likelihood of LOCA during inerting/de-inerting is very low.

   - If a LOCA does occur, these valves will receive a close signal long before the onset of fuel failure. Note that the onset of fuel failure is when the core is uncovered and reactor water level 3 (when ACS valves isolate) is 3.8 meters above the core.

   - In the event of a radioactivity leak during inerting/de-inerting, the radiation detectors at the purge and vent exhaust line will detect the condition and isolate the ACS containment isolation valves. Note that the exhaust radiation detectors are very sensitive and are set at a lower setpoint compared to the ones inside containment to have an effective early detection.

- For ACS, a more reliable isolation valve is necessary to ensure containment integrity. A fast closing valve may be less reliable than valves with moderate speed. Faster closing may cause damage to valve seats, resulting in poor closure tightness, which is essential to the prevention of contaminant release to the environment.

- The difference between 5 seconds and 20 seconds is considered to be insignificant.

Therefore, the risk is judged to be sufficiently small and that the 20-second closure time, is deemed sufficient and reliable. The USNRC has accepted this position in the FSER, NUREG 1503.

3. The following points are to be noted from PSAR Section 6.2.4.3.6, which are accepted by the USNRC relevant to meeting the intent of Criterion 54:

- The ACS containment purge system has redundant containment isolation valves each powered from independent electrical division. Powering the isolation valves from independent electrical division assures diversity, since the failure of one division will not affect the other valve.

- These valves are designed to ***fail in the closed position*** upon loss of air or loss of electric power to the pilot solenoid valve.

- With the exception of the makeup valves, all containment purge system containment isolation valves are in the closed position during normal reactor operation.

- The purge and vent valves are open ***only*** during the inerting and de-inerting modes.

- These valves are outside containment and accessible ***should manual actuation be required.***

Since this arrangement has adequate redundancy and independence, and is not vulnerable to common mode failures, it meets the intent of Criterion 54.

4. The PSAR Table 6.2-2d, Item A.5.a. (Basic specification, Number of filter train) will show "2" instead of "1." Also, Table 6.2-2d, Item A.5.b., "Component specification," will be revised to read "Component specification (per filter train)."

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 06-024

PSAR Sections: App. 6A

Question Date: May 4, 1998

PSAR Question:

1. To satisfy the fifth item (In-Place Testing Criteria) requirement, PSAR committed that the air flow distribution test of ANSI N510-1975 will be conducted during initial construction. Please clarify how this commitment can be realized ?

2. Please also clarify how the similar commitments such as page 6A-10, items (2) & (3) on activated carbon procurement and testing requirements (ANSI N509-1976) can be realized ?

Response:

1. PSAR Appendix 6A, Item 6A(5)(b) states that acceptance tests (including the air flow distribution test of ANSI N510-1975) will be "performed after completion of initial construction and after any system modifications or repair." These tests will be accomplished according to the methods of Section 8.3.2 of ANSI N510 - 1975.

2. Page 6A-10, item (2): Any supplier of activated charcoal for the SGT system will have to meet the requirements of Table 5.1 of ANSI N509 - 1976; this will be a requirement in the purchase specification.

   Page 6A-10, item (3): The filter trains will allow for easy removal of charcoal samples, which will then be laboratory tested in accordance with ASTM D3803, to meet the requirements of Table 2 of Reg. Guide 1.52. If the samples fail these tests, the charcoal will be replaced with a new batch.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: PSAR Typographical Errors

PSAR Sections: Ch. 6

Question Date: May 4, 1998

PSAR Question:

1. The Figures in Section 6.2.3.2 (System Design) of Chapter 6 such as Figures 1.2-2/1.2-2a/1.2-4, etc. were not issued yet so the descriptions of them should be of future tense to be consistent. This is a generic case and similar situations occur throughout the following sections.

2. There are many page number errors in the Table of Contents of Chapter 6 such as Table 6.2-7/6.2-8/6.2-9/6.2-10 and Figures 6.2-27/6.2-28. Please correct them.

Response:

1. Whereas English language is precise about the tenses, usage of present tense is acceptable in documents like the PSAR in which many sections are cross referenced and written independently. Since the PSAR was written using the SSAR as a starting point, the present tense was retained. Further, the PSAR will be a starting point for the FSAR. Sine the FSAR will be a complete document, the present tense will be appropriate for it. In order to reduce the errors in the tense of the FSAR during conversion from the PSAR, it is recommended that the present tense be retained in the PSAR in referencing the information to be provided in the FSAR.

2. The page number errors in Chapter 6 List of Tables and List of Figures will be corrected.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-001

PSAR Sections: Ch 7.2

Question Date: December 19, 1997

PSAR Question:

1. 10CFR 50.55a(h) requires that protection system meet IEEE Standard 279 or its revised editions. IEEE has issued revised standard--IEEE 603--for replacing IEEE 279 in 1991. Please revise the contents of PSAR Chapter 7 based on the requirements of IEEE Std 603 1991.
2. According to BTP HICB-14, the information to be reviewed consists of 11 software life cycle related plans. Please provide these plans.
3. GDC 29 requires "The protection system be designed to assure an extremely high probability of accomplishing their safety functions.." In Standard Review Plan Appendix 7.1A, a probability assessment is suggested as a method to determine whether GDC 29 has been met. Please provide necessary information (e.g., fault tree, event tree, reliability, etc.) such that a detail probability assessment for protection systems can be performed.

Response:

1. GE was originally prepared to remove all references to IEEE 279 in the PSAR based on a proposed rulemaking by the USNRC that would have permanently changed the reference in 10CFR50.55(a)(h) to IEEE 603. However. the rulemaking was withdrawn by the NRC due to adverse public comments (related only to domestic retrofit issues). An associated proposed rulemaking is still active and can be issued when the public comments are resolved. Since TPC is reluctant to make any changes without the rulemaking, GE does not plan to revise the PSAR until the situation is resolved. However, please note that IEEE 603 is referenced within Chapter 7 to discuss Engineered Safety Features (ESF) and is fully described as containing all elements of IEEE 279 plus additional material for command and sense features of ESF. Thus, the PSAR presently conforms to the use of IEEE 603 even while referring to the withdrawn IEEE 279.

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

Further Clarification to ROC-AEC's Comments:

Chapter 7 of the PSAR will be updated to address the requirements of IEEE Standard 603-1991 where currently IEEE 279-1971 is referenced. References to IEEE Standard 279-1971 will be removed except in those instances were it is appropriate to utilize it.

2. BTP HICB-14 recommends the contents of 11 software life cycle-related plans. Within Lungmen DCIS, the software development process is contained in three major development plans: (1) the Software Management Plan, (2) the Software Configuration Management Plan, (3) the Software Verification and Validation Plan. These three plans define: (1) the technical requirements for the job and (2) the programmatic requirements for the job. These plans will be provided to ROC-AEC.

   These three plans, plus two other supporting plans, embody all requirements outlined in the eleven plans of BTP-14. The additional plans are the Lungmen DCIS Essential Controls Project Plan and the Lungmen DCIS Essential Controls Validation Test Plan. These plans contain certain project planning details involving schedule and budget that are not covered by the other three plans. These two plans will be provided to ROC-AEC when available.

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

Further Clarification to ROC-AEC's Comments:

The NRC's Branch Technical Position HICB-14 (BTP-14) outlines the many activities to be considered when constructing a software development program for *safety-related* software. As an example format, BTP-14 divides these activities into 11 separate software development plans (note that this is only a sample format and that eleven separate plans are not specifically required). The only requirement is that the essential elements of each of the 11 development groups has been addressed and documented. The required development activities for Lungmen DCIS safety-related software are documented in the following plans :

1. Software Management Plan (SMP): Target Schedule: April 30, 1998.
2. Software Configuration Management Plan (SCMP): Target Schedule: April 30, 1998.
3. Software Verification and Validation Plan (SVVP): Target Schedule: April 30, 1998.
4. Software Development Project Plan (PP): Target Schedule: April 30, 1998.
5. Software Safety Plan (SSP): Target Schedule: October, 30, 1998
6. Software Validation Test Plan (SVTP): Target Schedule: March, 1, 1999

   Additionally, certain details of the software V&V process are described in the:
7. Independent Verification and Validation Plan (IVVP): Target Schedule: June, 1, 1998. This plan will be produced by the Independent V&V organization function of the Independent Review Team (IRT).

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

3. Reactivity control in the ABWR includes diverse systems with redundant components to enhance the reliability of reactor shutdown when needed.. Top level fault trees for the two diverse reactivity control mechanisms (by control rod insertion, and liquid boron injection) are presented in the ABWR SSAR Figure 19D.6-16a, p. 19D.6-195. Figure 19D.6-16b shows the contribution of RPS to the reactivity control function. Notice that failure of the RPS is necessary, but not sufficient for the failure of control rod insertion since the ARI has to also fail. Figure 19D.6-24 of the SSAR presents the detailed fault tree of the RPS. Section 19N of the SSAR contains analysis of the related issue of common-cause failure of the multiplex equipment.

Lungmen-specific fault trees for the reactivity control function have been developed for some systems in Section AA.2.4 of the PSAR Appendix A. These fault trees will be updated and completed to reflect the final design in the FSAR.

Further Clarification to ROC-AEC's Comments:

Software failure effects within the digital protection system equipment were not explicitly considered in the PRA study because it is not possible at present to quantify software design errors in complex programs, particularly on software that is not yet written, which is the case for the protection system (Safety System Logic and Control and Essential Multiplexing System). Failures of memory chips in the SSLC and EMS controllers that contain software control programs are considered hardware failures and are calculated as part of the mean time between failure (MTBF) of each controller. This overall MTBF was used for the PRA.

GE has established an extensive software development process (see Question 2 of the Track Number) for producing safety-related software. This process includes a safety analysis of the software at each stage of development plus validation testing of the finished product against design requirements to ensure the lowest probability of error. However, GE agrees with the following NRC position as stated in Branch Technical Position (BTP) HICB-19, Section B, of the revised Standard Review Plan (SRP), Chapter 7:

"...software design errors are a credible source of common-mode failures. Software cannot be proven to be error-free, and therefore is considered susceptible to common-mode failures because identical copies of the software are present in redundant channels of safety-related systems. To defend against potential common-mode failures, the Staff considers high quality, defense-in-depth, and diversity to be key elements in digital system design. High-quality software and hardware reduces failure probability. However, despite high quality of design, software errors may still defeat safety functions in redundant, safety-related channels. Therefore, ... the Staff requires that the applicant/licensee perform a D-in-D&D assessment of the proposed digital I&C system to demonstrate that vulnerabilities to common-mode failures have been adequately addressed. In this assessment, the applicant/licensee should analyze design basis events (as identified in the safety analysis report). If a postulated common-mode failure could disable a safety function that is required to respond to the design basis event being analyzed, then a diverse means of effective response (with documented basis) is necessary. The diverse means may be a non-safety system, automatic, or manual if the system is of sufficient quality to perform the necessary function under the associated event conditions and within the required time."

As stated in a previous response (Track Number 07-011), the required diversity and defense-in-depth analysis has been performed for the ABWR and, furthermore, will be updated to include Lungmen-unique features.

The safety analyses performed during software development will establish a qualitative reliability estimate that is endorsed by the NRC as follows in the SRP, Chapter 7.1, Section II, Item 6:

"...The Staff does not endorse the concept of quantitative reliability goals as a sole means of meeting the NRC's regulations for the reliability of digital computers used in safety systems. The NRC staff's acceptance of the reliability of the computer system is based on deterministic criteria for both the hardware and software rather than on quantitative reliability goals. ... Nevertheless, qualitative reliability estimation using a combination of analysis, testing, and operating experience can provide an added level of confidence in a system's reliable performance. Qualitative estimation of software reliability should address the fact that software failures that are not the consequence of hardware failures are caused by design errors and, therefore, do not follow the random failure behavior used for hardware reliability."

Item 8 of the same section addresses emerging technologies which might aid in establishing quantitative software reliability. However, the NRC will not accept use of these methods to demonstrate compliance with the fundamental acceptance criteria.

As discussed above, GE has committed to produce highly-reliable safety-related software. GE's development program is in compliance with SRP criteria.

Data that relates core melt probability to either RPS or ESF software failure is not available at this time. In addition, software failure rates that are specific to either RPS and/or ESF are not available at this time. Both types of information will not be available until after the specific architecture for RPS and ESF have been defined and probabilistic risk analyses have been performed. These analyses are not scheduled to be performed until the FSAR stage.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-002

PSAR Sections: Ch 7.2

Question Date: December 29, 1997

PSAR Question:

1. Compared with Kuosheng, the trip signals in the RPS for Lungmen do not include "main steam line high radiation" or "RPV high water level" trip signals. Why are they removed ? Is there replacement protection design? Please clarify.

2. Suppression Pool serves as heat sink. When heat is added (not from SRV discharge due to RPV high pressure trip), the temperature rise will be gradual and when it reaches certain level the cooling system will start and lower the temperature. Compared with Kuosheng, which does not have igh Suppression Pool Temp" trip signal, Lungmen has included such signal in its RPS. Please explain the design philosophy and why the necessity for such trip signal.

3. Is the Lungmen RPS expandable? Please explain.

Response:

1. The elimination of the automatic reactor shutdown of the Main Steam Line Radiation Monitor (MSLRM) is based on the GE safety analysis contained 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". NEDO-31400A evaluated and justified the removal of the above MSL high radiation trips and has been reviewed and accepted by the US NRC. Many GE BWRs have now removed the reactor scram and vessel isolation trip functions as a result of this evaluation.

   The elimination of the automatic reactor scram function of the MSLRM will reduce the potential for unnecessary reactor shutdowns

caused by the spurious actuation of the MSLRM trips and will increase the plant operational flexibility. This change also addresses TPC Bid Specification. Section 3.4.8.2.1(2) which requires that the number of the RPS trips be optimized.
The MSL high radiation trip signal, as utilized by the Reactor Protection System, was not removed from the ABWR SSAR text due to the date at which NEDO-31400A was finally approved by the USNRC. GE felt that incorporating the removal of this trip into the SSAR during the period at which the SSAR was being actively reviewed by the USNRC could potentially interfere with the timely certification of the SSAR. Therefore, even though the USNRC had ultimately approved the removal of the trip, GE had decided not to incorporate the change in order not to jeopardize the ultimate issuance of the Final Safety Evaluation Report (FSER).

The three conditions in NEDO-31400A that the USNRC stipulated for successful implementation of the elimination of the MSLRM trip that shuts down the reactor and closes the Main Steam Isolation Valves, are addressed as follows:

- Lungmen NPS unique input values for determining the acceptability of the change were compared against the values used in generic analysis and were found to be acceptable.
- As part of the operating procedures, TPC will develop and incorporate information providing guidance to operators on measures to take upon detection of high radiation in the main steam lines to limit both occupational doses and environmental releases.
- The alarm for the Main Steam Line radiation monitor will be established at 1.5 times the full power radiation background in the main steam line tunnel. In addition, TPC will commit to promptly sample the reactor coolant for possible contamination and determine the need for additional corrective actions if either, or both, the main steam line radiation monitor and offgas pretreatment radiation monitor exceed their alarm setpoints.

Additional information on the reason and background of the MSLRM change can be found in NEDO-31400A.

The design of the GE ABWR, as documented in the Standard Safety Analysis Report (SSAR), does not include an RPS automatic reactor trip on "RPV high water level". The design of the Lungmen NPS RPS, as described in the section 7.2 of the PSAR, does not propose any changes regarding RPV water level and is the same design as defined in the ABWR SSAR.

Chapter 15 of the Lungmen NPS PSAR , "Accident and Analysis", evaluates the events that could affect the reactor coolant inventory. Please refer to the following sections of Lungmen NPS PSAR:

- Sections 15.5, "Increase in Reactor Coolant Inventory" (inadvertent startup of HPCF).
- Section 15.1, "Decrease in Reactor Coolant Temperature" (Feedwater Control Failure Maximum Demand will result in both reactor coolant temperature decreasing and reactor coolant inventory increasing).

In the ABWR SSAR and for the Lungmen NPS design, the design is based upon having the RPV high water level trip function being implemented in different microprocesser channels than those used for the Feedwater Control System level control algorithms.

The US NRC accepted this design approach and required this function to be implemented in safety system logic. Kuosheng (and most BWR/6 plants) included the high water level SCRAM in the Reactor Protection System trip logic to provide for improved reliability of the desired high water level shutdown of the reactor. The turbine trip and feedwater pump trip logic was still implemented in separate feedwater control system logic. Redundancy was achieved for this logic by using three separate analog trip units (or equivalent) and performing 2-out-of-3 voting for the turbine trip/feedwater pump trip initiation logic.

For the Lungmen NPS, it was decided that the Steam Bypass and Pressure Control System (C85) system would include the logic that initiates trips of the Main Turbine and the feedwater pumps (with initiation of automatic closure of the associated main discharge valves) on high RPV water level (i.e., level 8).

It should be noted that GE ABWR major non-safety control system designs, unlike the original conventional BWR plant (e.g., Kuosheng) designs, utilize triplicated fault tolerant digital controllers for the key plant control systems (e.g. Feedwater Control System, Steam Bypass and Control System).

The triplicated channel control logic of the C85 system is separate and diverse from the Feedwater Control System (C31) logic. This requirement has been imposed so that a Feedwater Control failure can not affect the Steam Bypass and Pressure Control System ability to initiate the associated trip functions at RPV level 8. With this approach, a highly reliable Level 8 trip function is included in the Lungmen design, without having to use the RPS for this high water level trip function.

For the US ABWR SSAR and the Lungmen NPS design, GE and the US NRC agreed that there was no requirement to include the high reactor water level trip function logic in safety system logic. So, by taking advantage of the triplicated controllers in the ABWR design, as previously described, there was no need to include the Level 8 trip in the RPS SCRAM functions. This simplifies the RPS design while still providing for highly reliable mitigation actions when Level 8 is reached.

Also, as Lungmen has turbine trip without scram capability (provided the fast opening function of the steam bypass valves works properly), the Lungmen NPS RPS design assures that no SCRAM is required when the Level 8 turbine trip is initiated and the fast opening function of the turbine bypass valves is operable. However, for the case of normal power operation, as all the feedwater pumps are tripped, the reactor will be SCRAMed when the reactor water level eventually decreases to the Level 3 condition. But, there is no requirement for a direct SCRAM on high water level conditions for providing water level transient mitigation. The turbine trip and feedwater pump trip functions provide the required mitigation for these type of transients.

No changes to the PSAR will be done as a result of the responses to Question 1.

2. High Temperature in the Suppression Pool will occurs upon an inadvertent opening of an SRV. To mitigate this event without operator intervention in the Lungmen NPS, an automatic reactor scram on high Suppression Pool Temperature is provided. It should be noted that, in the case of an inadvertent Safety Relief Valve opening, it is possible for the suppression pool temperature (even with start of the cooling system) to rise to the point at which reactor trip would be required. Table 15.1-8 of the Lungmen NPS PSAR, "Sequence of Events for Inadvertent Safety/Relief Valve Opening" provides the following time sequence:

   - 0.5 Sec. after inadvertent SRV opening, relief flow reaches full flow

   - 750 Sec. after inadvertent SRV opening, Suppression Pool Temperature reaches setpoint for the suppression pool cooling function to initiate.

   - 1200 Sec. after inadvertent SRV opening, Suppression Pool Temperature reaches setpoint for the automatic reactor scram to initiate.

   The Suppression Pool High Temperature (SPHT) scram trip function was added to the Lungmen NPS design based on operator response requirements, i.e., no operator actions for 30 minutes after a transient and the ABWR objectives for greater simplicity of operation. The SPHT Scram design is not a USNRC requirement for BWRs. Standard Emergency Operating Procedures and Plant Technical Specifications would require that the operator take manual actions to control Suppression Pool high temperature. If those manual actions were not taken, the automatic systems, such as RHR Suppression Pool Cooling and RPS Scram on SPHT, would preclude exceeding the design parameters after a transient. As shown in PSAR Section 15.1.4 Inadvertent Safety/Relief Valve Opening transient analysis, with no assumed operator action, the transient is terminated by automatic suppression pool cooling initiation and reactor scram as the suppression pool temperature eventually reaches the trip set point.

The SPHT Scram design was not implemented on the currently operating ABWRs as the design improvement came after final design of the operating ABWRs.

No changes to the PSAR will be done as a result of the responses to Question 2.

3. The RPS design , as described in Lungmen PSAR section 7.2, is the accident prevention arm of the ABWR' plant protection systems. It meets all the established requirements on plant safety, availability, and maintainability.

   The architecture of the system is made flexible and is able to tolerate future modifications, such as addition of new manual or automatic tests, bypasses or permissive for bypasses, and even addition of new trips (although it would be highly improbable). We should note that, no part of the Class 1E RPS can be changed without the appropriate licensing effort. To add new trip parameters, for example, requires software changes to the embedded control software and may require additional hardware changes, depending on the change proposed. Thus additional software V&V and QA (and hardware requalification, if applicable) would be required.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-003

PSAR Sections: Section 7.2

Question Date: January 14, 1997

PSAR Question:

The description on the commitments of code compliance in Sectons 7.2 to 7.9 is not complete and inconsistent with Table 7.1-2. Taking Reactor Protection System as an example, Table 7.1-2 listed that it will comply with BTP HICB 14, 17, 18, 19 and 21 etc., but in Section 7.2.2.2.4, "Conformance to Branch Technical Positions", no such commitments were made. Similar situations occur in Sections 7.3 to 7.9. Please correct it.

PSAR Response:

Please note that Table 7.1-2 contains asterisks (*) on the "Applicable Criteria" row for a number of identified Regulatory Guides and HICB , including those indicated in the given example. The associated footnote on Table 7.1-2 indicates these criteria are addressed in conjunction with the SSLC, which is the logic and control interface with the identified systems (also, see Table 7.1-1). The reader is then referred to Subsections 7.1.2.11 through 7.1.2.13 for these conformance commitments.

Note that the NMS does not utilize the SSLC for its logic and control interface; therefore, the criteria applicable to the NMS are addressed independently in Subsection 7.7.2.6.

In summary, all criteria in Table 7.1-2 are addressed in the PSAR. The items without the asterisks are addressed in the individual system analysis sections for those systems having an " " on the table. Those criteria with the asterisks are addressed generically in Subsections 7.1.2.11 through 7.1.2.13, but may have additional information, as appropriate for some systems, in the individual analysis sections.

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

A reference to 7.1.2.11 through 7.1.2.13 will be added within the applicable systems analysis sections of the PSAR.

The Chapter 7 PSAR text and Table 7.1-2 were checked for consistent inclusion of all codes and standards required per the SRP when the PSAR was issued. However, GE will perform an additional check during the next update of the chapter.

In order to further clarify the format for Section 7.5.2.1 on page 7.5-7, the following is provided: Consistent with the ABWR Certified Design, the Lungmen NPS post accident monitoring (PAM) is not a designated "system" of itself. Rather, PAM is accomplished by assimilation of a large set of discrete data, as defined by Regulatory Guide 1.97, which is gathered from the variables and parameters of many existing systems. Sections 7.5.1.1 and 7.5.2.1 of the PSAR provide an assessment of Regulatory Guide 1.97, which identify how the variables available from existing systems fulfill the PAM requirements. Therefore, Regulatory Guide 1.97 is addressed in detail for the PAM.

Since the PAM is not a system in the same sense as the others, a subsection for "General Functional Requirements" would have no meaning, and the "Specific Regulatory Requirements Conformance" is already covered is Section 7.5.2.1. This is consistent with the format in the ABWR SSAR, which has been approved by the NRC.

However, in order to clarify this concept, the word "system" will be deleted from the PAM titles in Section 7.5, consistent with its designation as "Post Accident Monitoring" on Table 7.2-1.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-004

PSAR Sections: Ch 7.4.2.2 Remote Shutdown System

Question Date: January 7, 1998

PSAR Question:

Since the Remote Shutdown panel or facilities are powered from class 1E power system, they should follow the requirements of RG 1.153. But in PSAR page 7.4-21 the R.G. listed were not consistent with Table 7.1-2 in page 7.1-57. Please explain.

PSAR Response:

Agreed. PSAR Section 7.4.2.2.2 (3) (page 7.4-21) will be revised to include RG 1.153. This revision will provide consistency between PSAR Section 7.4.2.2.2 (3) and PSAR Table 7.1-2 (page 7.1-57).

Specifically, PSAR Section 7.4.2.2.2 (3) should be revised to include:

(d) RG 1.153 - Criteria for Power, Instrumentation, and Control Portions of Safety Systems

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-005

PSAR Sections: Ch 7.7.1.1 Main Steam System

Question Date: December 18, 1997

PSAR Question:

1. The reference heights of the RPV such as RVZ, IZ and TAF should have definite numbers associated with them and their definition should be made clear during design.

2. The different kinds of water level instruments should have their range of measurement, reference zero point and calibration environment clearly specified with numbers (height). The scram logic in design should also be explained in detail and drawings attached.

3. Please specify the quantity of water level instruments clearly.

4. Please explain in detail the quantity and functions of the RPV pressure instruments in design.

5. Please use numbers to clearly identify the setpoints for those RPV water level and pressure instruments that have a scram function.

6. Please explain where the pressure switch on the RPV head is located and its alarm setpoint.

PSAR Response:

1. IZ, instrument zero for the water level range, is at the top of the active fuel. RVZ and TAF reference dimensions will be added to Figure 7.7-1of the PSAR.

2. Analytical limit of water level 1 from vessel zero is 920.2 cm (362.3 in) and from TAF is 5.2 cm (2.1 in). Analytical limit of water level 1.5 from vessel zero is 1003.6 cm (395.1 in) and from TAF is 88.6 cm (34.9 in). Analytical limit of water level 2 from vessel zero is 1148.3

cm (452.1 in) and from TAF is 233.3 cm (91.9 in). Analytical limit of water level 3 from vessel zero is 1279.7 cm (503.8 in) and from TAF is 364.7 cm (143.6 in). Analytical limit of water level 8 from vessel zero is 1395.4 cm (549.4 in) and from TAF is 480.4 cm (189.2 in). The Scram logic design is described in PSAR section 7.2.1.1.4.3.
These design details are provided in response to this question, and will also be provided in the FSAR. However, no change to the PSAR will be made as a result of the response to this Question.

3. As shown in Figure 5.1-3 (sheet 4), there are 9 narrow range RPV water level transmitters. They are: 1B21-LT-0013A,B,C; 1B21-LT-0014A,B; 1B21-LT-0016A,B,C,D. There is one shutdown range RPV water level transmitter, i.e., 1B21-LT-0015. There are 9 wide range RPV water level transmitters, i.e., 1B21-LT-0019A,B,C,D,E,F,G,H and 1B21-LT-0020.

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

4. The RPV is equipped with 11 pressure transmitters. They are: 1B21-PT-0009A,B,C,D; 1B21-PT-0011 A,B,C; 1B21-PT-0012A,B,C; 1B21-PT-0026. The MSIV actuators have 4 pressure transmitters; i.e., 1B21-PT-0040A,B,C,D. These pressure transmitters monitor RPV conditions and the MSIV accumulator pneumatic pressure. Pump deck differential pressure transmitters are 1B21-PDT-0029A,B,C,D and core plate differential transmitters are 1B21-PDT-0031A,B,C,D. Information provided by these PDTs is used in the Neutron Monitoring System biasing calculations.

   These design details are provided in response to this question, and will also be provided in the FSAR. However, no change to the PSAR will be made as a result of the response to this Question.

5. Reactor scram is initiated by reactor vessel low water level (Level 3). Analytical limit of water level 3 from vessel zero is 1279.7 cm (503.8 in) and from TAF is 364.7 cm (143.6 in). Drywell pressure scram initiation analytic limit is 13.8 kPaG (2.0 psig). Reactor scram is also initiated upon the condition of high reactor vessel dome pressure The

minimum plant safety analytical limit for the high pressure scram is 7.62 MpaG (1105 psig).

These design details are provided in response to this question, and will also be provided in the FSAR. However, no change to the PSAR will be made as a result of the response to this Question.

6. A pressure switch is not provided on the RPV head. However, as stated in PSAR section 5.2.5.2.1 (7) " A single channel of pressure monitoring is provided for measurement and control room indication of pressure between the inner and outer reactor head flange seals....". No change to the PSAR will be made as a result of the response to this Question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-006

PSAR Sections: Ch 7.9.1.1.1.1 and 7.9.1.1.2.1

Question Date: January 19, 1998

PSAR Question:

1. The titles of sections 7.9.1.1.1.1 and 7.9.1.1.2.1 are "NEMS System Interfaces" and "EMS System Interface" respectively but only the system names that have interfaces with NEMS/EMS were listed. Please explain which kind of signals from each system were sent to NEMS/EMS and if the scan rate of NEMS/EMS to the various signals will be different depending on the signal's change rate, normal/transient states, etc. Also, please explain how NEMS/EMS does its best sampling of the various signals from the different systems to avoid signals losing their fidelity.

2. Each of the system which interfaces with NEMS/EMS has a code number, such as Main Steam System (B21). Is B21 the system code? Have the coding rules for Lungmen system, equipment and components been decided to avoid the confusion in operation and maintenance due to two coding rules for system and equipment at the Second NPS?

PSAR Response:

1. Those signals (flux, flow, status, pressure, levels, etc.) necessary for the operator to properly control, monitor, and alarm the respective system are sent to NEMS/EMS.

   The scan rates will be different for each signal type; for example temperatures are scanned more slowly than pressures. But all signals are scanned at rates suitable for post transient event analysis.

   The NEMS/EMS are not system oriented; signals from many systems (appropriately separated by division) can be mixed on the same RMU. Each input to the RMU is individually isolated and sampled without interaction / interference with any other signal on the RMU. There will be no loss of signal fidelity in this process.

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

2. B21 is the system code for the Main Steam System. The coding rules for Lungmen NPS systems, equipment and components have been established so that there will be only one identification system used throughout the plant. This coding system will be used in the PSAR, FSAR and in the operation and maintenance of the plant.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-007

PSAR Sections: Ch 7.9.1.1.1.4 NEMS Equipment

Question Date: January 20, 1998

PSAR Question:

Please explain which non-safety signals from NEMS will be passed to EMS and how the interface is done? (The original paragraph is listed here for reference : "...pass some non-safety signals to the safety system for display purposes.")

PSAR Response:

NEMS will **NOT** pass any information to EMS. The referenced statement will be deleted in the next revision of the PSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-008

PSAR Sections: Ch 7.9 Data Communication System

Question Date: January 20, 1998

PSAR Question:

Networking and Multiplexing are the communication media for large amount of two way signals between the field and the control room. So their failure will have far more consequences than conventional analog signal systems. Please explain separately the Failure Mode and Effect Analysis of EMS and NEMS.

PSAR Response:

The EMS Failure Mode and Effect Analysis is contained in PSAR Section 15B.4 and Table 15B-3, and Appendix AI.6 addresses the common-mode failure of multiplex equipment.

The NEMS is diverse in both hardware and software from the EMS. As the name implies, the NEMS is non-Class 1E system, so a Failure Mode and Effect Analysis was not performed.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-009

PSAR Sections: Ch 7.9 Data Communication System

Question Date: January 21, 1998

PSAR Question:

This section covers networking and multiplexing which are indispensable systems for digitization so the new edition of SRP has included it as one of the more important new sections. However, in the PSAR there are hardly any drawings to further the explanation and compared with GESSAR which provided a lot of drawings, the PSAR is not quite complete. So please at least provide the drawings with explanations for the sections below:

1. 7.9.1.1.1 NEMS System Description
2. 7.9.1.1.1.3 NEMS Power Sources
3. 7.9.1.1.2 EMS System Description
4. 7.9.1.1.2.3 EMS Power Sources

PSAR Response:

1. The following text and Figure 7.9-1 will be appended to Section 7.9.1.1.1

"A sketch showing the general configuration of NEMS is shown in Figure 7.9-1. The intent of this figure is to show general configuration of NEMS interfacing with the related systems. Information relating to these interfacing systems are shown only for the purpose of illustration and delineating the interfaces. The NEMS scope is marked by the double arrow.

The NEMS main "backbone" consists of redundant high speed fiber optic data paths from the RMUs in the field to the dual or triply redundant controllers located in the CB. NEMS has the following major equipments:

1. Remote Multiplexing Units (RMU) - The RMUs are located throughout the various buildings (RB, CB, Turbine Building, radwaste, switchyard, pumphouses and Auxiliary Fuel Building etc.). The RMUs acquire signals from remote process sensors and actuate valves and motors drives in return. The RMUs connect directly to the dual or triply redundant controllers in the CB via redundant fiber optic links.
2. Dual/Triple redundant controllers - Each RMU (or in some cases groups of RMUs) connects to dual/triple redundant controllers located in CB cabinets. These controllers perform the functions of interfacing the RMUs to the Non-Class 1E Fast Ethernet, and it also performs batch and continuous control functions of Non-Class 1E process controls.
3. Non-Class 1E Fast Ethernet - The Fast Ethernet provides the media for the redundant controllers to exchange information. The Plant Computer System and various VDUs are also connected to the Fast Ethernet to access information.
4. Data Gateways - The data gateway provides the isolation to the safety-related controllers. Safety-related signals are received by the Plant Computer System through these data gateways for historical and display purposes."

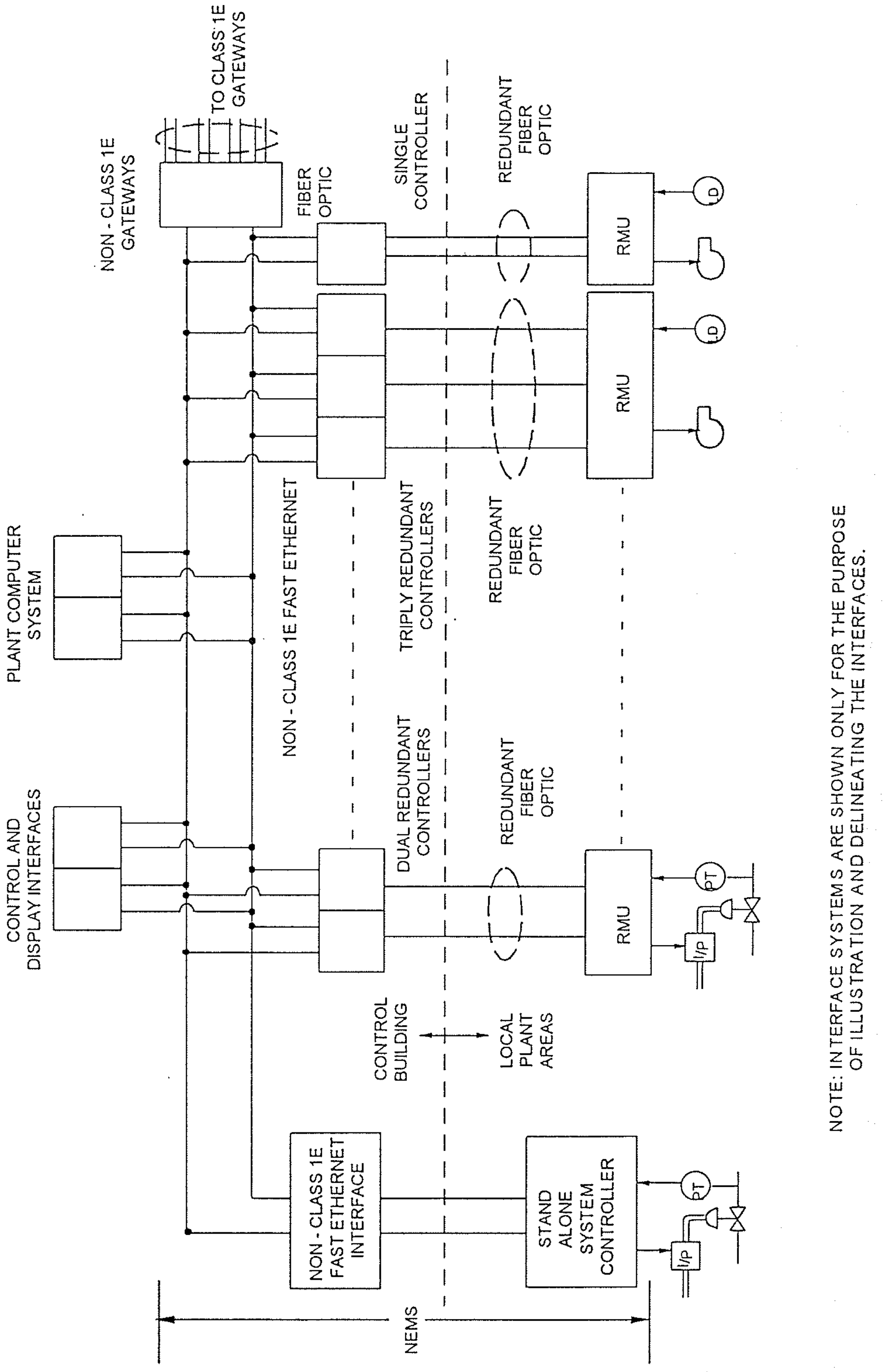


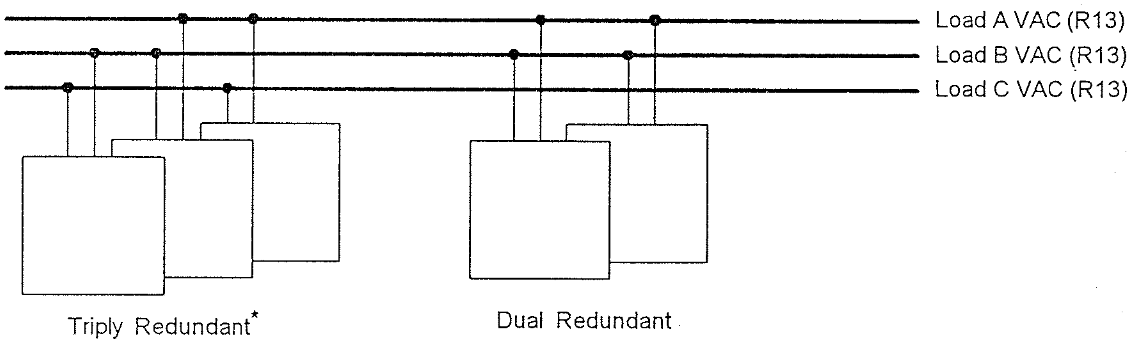
Figure 7.9-1 Non-Essential Multiplexing System Configuration

2. The following text will replace 7.9.1.1.1.3 and Figure 7.9-2 will be appended to this section.

"7.9.1.1.1.3 NEMS Power Sources

The NEMS is powered by two or three separate Non-Class 1E distribution panels from the Non-Class 1E 120VAC (R13). This redundancy allows the NEMS to supply duplicated logic functions such that any single failure in the system power supplies will not cause the loss of the validated outputs to the interfacing actuators and to monitors and displays.

The power sources automatically switch over upon failure of one power source or power supply module. For the triplicated controllers, three sources of non-safety-related AC power are provide with two sources used by each channel. Figure 7.9-2 provides the typical power distribution for dual and triply redundant connections.



* Depending on power supply arrangement, there may only be one power feed.

Figure 7.9-2 NEMS Power Sources"

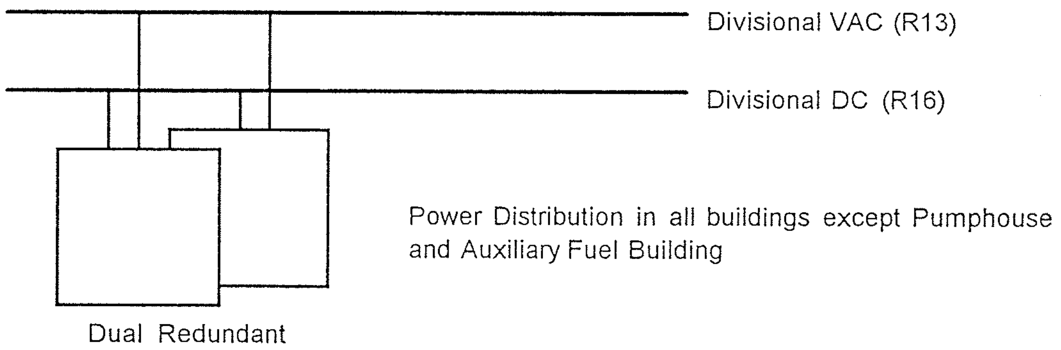
3. EMS system configuration has been changed since the issuance of PSAR. GE has been in discussion with TPC about this change. However, no final approval has been received. When the final decision is made, a detailed description and Figure 7.9-3 will be provided regarding the EMS system configuration.

4. The following text will replace 7.9.1.1.2.3 and figure 7.9-4 will be appended to this section.

"7.9.1.1.2.3 EMS Power Sources

The EMS receives it primary power from the divisional Class 1E 125VDC Power Supply (R16), and each division of DC powers the associated division of EMS. The backup power source for EMS is the Class 1E 120VAC (R13). Each division of VAC powers the associated division of EMS if DC fails. Figure 7.9-4 provides the typical power distribution for EMS.

Those RMUs that are located in the safety-related pumphouses and AFB are powered by Class 1E 120VAC (R13) and Class 1E Low Voltage Distribution System (LVD) 120VAC (R12), and the output from these two power supplies is auctioneered. The divisional LVD (R12) is generated by the divisional 480V power through a step down transformer.



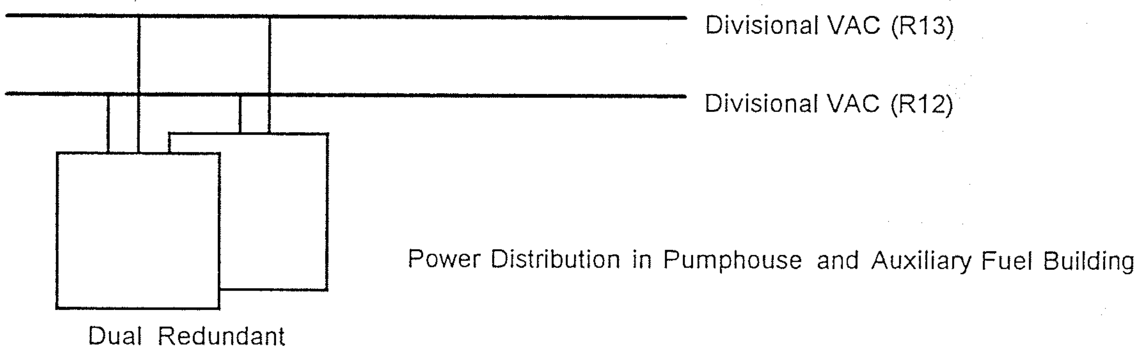


Figure 7.9-4 EMS Power Sources"

RESPONSES TO ROC-AEC's PSAR QUESTIONS

ROCAEC Review Comments:

The response to Item 3 of 07-009 says: "EMS system configuration has been changed since the issuance of PSAR. GE has been in discussion with TPC about this change. However, no final approval has been received. When the final decision is made, a detailed description and Figure 7.9-3 will be provided regarding the EMS system configuration." Response with open action is not acceptable.

Further Clarifications:

A detailed description, consistent with final approval by TPC, is provided below:

The new response to Item 3 is presented below.

3. PSAR Section 7.9 will be revised as follows:

a) Section 7.9.1.1.2, Paragraph 2 is revised to read as follows:

7.9.1.1.2 **EMS System Description**

The EMS acquires both analog and digital signals from remote process sensors and discrete monitors located within a plant and multiplexes the signals by redundant fiber optic lines to the MCR for transmission to the Reactor Protection System (RPS) and Engineered Safety Features (ESF) portions of Safety System Logic & Control (SSLC), and other divisional controllers. Within each division of EMS there are two separate networks, one for RPS/MSIV (see Figure 7.9-3) and one for ESF (see Figure 7.9-3a), which are independent from one another.

b) Section 7.9.1.1.2.4, is revised to read as follows:

7.9.1.1.2.4 EMS Equipment

Each EMS division comprises remote multiplexing units (RMUs), and the redundant fiber optic signal transmission path. EMS contains a minimum of one main control room multiplexing unit (MCR MUX) plus other RMUs located in each of the four divisions in the Control Building (CB) and several RMUs located in each of the emergency electrical equipment rooms in the Reactor Building (RB), CB and safety-related pumphouse. The number of RMUs varies per division as the quantity of sensor inputs and outputs varies. The interface linking SSLC with EMS is contained in the digital trip modules (DTMs) of RPS/MSIV and ESF. The DTMs are dual to preserve the redundancy of each channel throughout the control loop.

c) Section 7.9.1.1.2.5, is revised to read as follows:

7.9.1.1.2.5 EMS Testability

No regular maintenance is required for EMS. However, continuous self-diagnostics in each RMU and DTM interface monitors the state of the system and each module. The system continues operating via the redundant channel after detecting an equipment functional failure to preserve the remainder of the network. Fault conditions that result in detected failures are alarmed in the MCR and are traceable to the lowest replaceable module.

Periodic surveillance, using off-line tests with simulated input signals, is used to verify the overall system integrity. Segments of EMS can be tested on-line when portions of SSLC are bypassed. The analog-to-digital (A/D) converters (also, digital-to-analog (D/A) converters, if used) in the RMUs are the only components requiring periodic calibration. This calibration will be performed automatically by injecting an accurate reference source, monitoring the results, and adjusting the zero and gain parameters.

d) Section 7.9.1.1.2.6, is revised to read as follows:

7.9.1.1.2.6 EMS Environmental Considerations

All components of EMS are located in controlled-environment, safety-related areas in the RB, CB, and safety-related pumphouse and qualified for their local environment. No components or fiber optic cables are located in the primary containment, or in high radiation areas. All signals from within these areas are hardwired with copper cable to the RMUs. The RMUs located in the emergency electrical equipment rooms in the RB and CB are cooled by the safety-related Reactor Building and Control Building HVAC Systems (RBHV & CBHV). The EMS interfaces located in the MCR area in the CB are cooled by the redundant Control Building HVAC System (CBHV).

e) Section 7.9.1.1.2.7, is revised to read as follows:

7.9.1.1.2.7 EMS Operational Considerations

EMS operates continuously in all modes of plant operation to support the data transmission requirements of the various interfacing systems. Data is continuously scanned and refreshed throughout the system. The EMS does not require operator intervention except to enforce a bypass condition. Power On/Off detection circuits are included in all active system components so that transient actuation signals are not generated under either condition. The EMS automatically restarts and establishes data transmission when power is applied to any segment of the system. Control and initiation outputs to driven equipment are held inactive until normal transmission is established, as monitored by the self-diagnostic software of the MUX controllers (RMUs and DTM interfaces).

In normal mode, the two fibers from an RMU to a DTM operate independently. When a network device fails or a fiber breaks, operation continues automatically, without operator intervention. In the event that a channel failure occurs, the network alarms in the MCR, indicating that a MUX failure has occurred, identifies which of the components has failed, and signifies which channel is operating. The failed segments of the channel are isolated from the operating segments and can be repaired and returned to service while on-line.

f) Section 7.9.1.1.2.8, is revised to read as follows:

7.9.1.1.2.8 EMS Operator Information

The following EMS displays and alarms are provided:

(1) MCR Alarms

(a) EMS Division I inoperative

(b) EMS Division II inoperative

(c) EMS Division III inoperative

(d) EMS Division IV inoperative

(2) Main Control Console Indications on Flat Displays:

(a) EMS Division I in degraded mode [a channel failure has occurred in a network segment or a faulty port has been removed from the system]

(b) EMS Division II in degraded mode

(c) EMS Division III in degraded mode

(d) EMS Division IV in degraded mode

(e) RMU failure (per unit and per division)

(f) RMU/DTM interface failure (per unit and per division)

g) Section 7.9.2.1.4.4, paragraph (4), is revised to read as follows:

7.9.2.1.4.4 Regulatory Guides

(4) Regulatory Guide 1.75-*Physical Independence of Electric Systems.*

The EMS complies with the criteria set forth in IEEE-279, Paragraph 4.6, and Regulatory Guide 1.75, which endorses IEEE-384. Class 1E circuits and Class 1E- associated circuits are identified and separated from redundant and non-Class 1E circuits. Isolation devices are provided in the design where an interface exists between redundant Class 1E divisions and between non-Class 1E and Class 1E or Class 1E-associated circuits. Independence and separation of safety-related systems is discussed in Subsections 8.3.1.3 and 8.3.1.4. Physical and electrical independence of the instrumentation devices of the system is provided by channel independence for sensors exposed to each process variable. Separate and independent raceways are routed from each device to the respective divisional remote multiplexing units (RMUs). Signals between redundant EMS divisions are electrically and physically isolated by fiber optic cables.

h) Section 7.9.2.1.4.4, paragraph (7), is revised to read as follows:

(7) Regulatory Guide 1.152-*Criteria for Programmable Digital Computer System Software in Safety-Related Systems.*

Software for EMS operation is developed and qualified for safety-related use together with the RMU hardware according to the software management plans described in Subsection 7.1.2.1.3 for SSLC which conform to ANSI/IEEE-ANS-7-4.3.2. Software based control programs are embedded as firmware, programmed into read-only memory, in the controller hardware and are not intended to be changed during operation.

i) Figures 7.9-3 and 7.9-3a as shown below will be appended to the end of Section 7.9 with the other figures in this Section.

j)

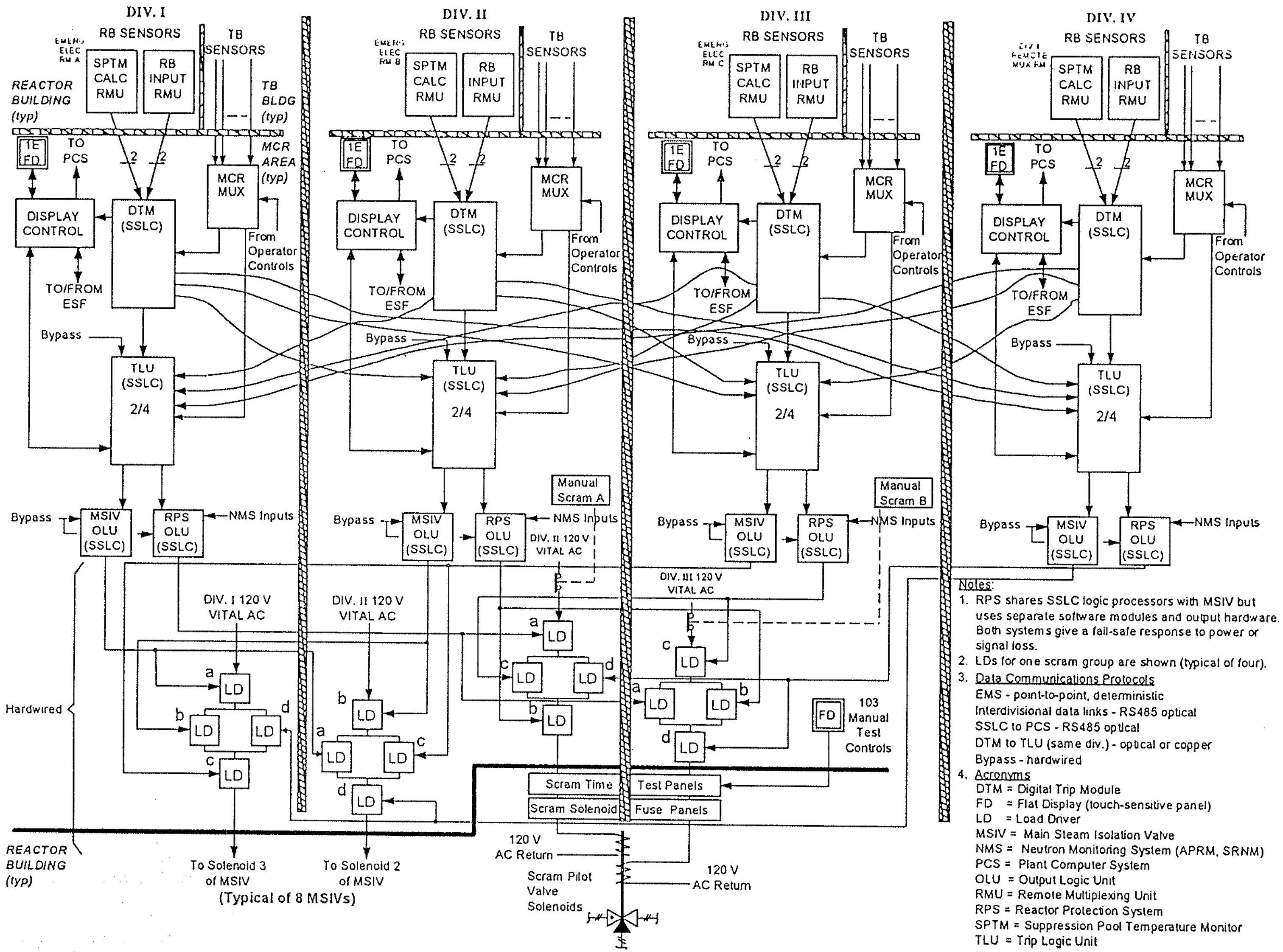
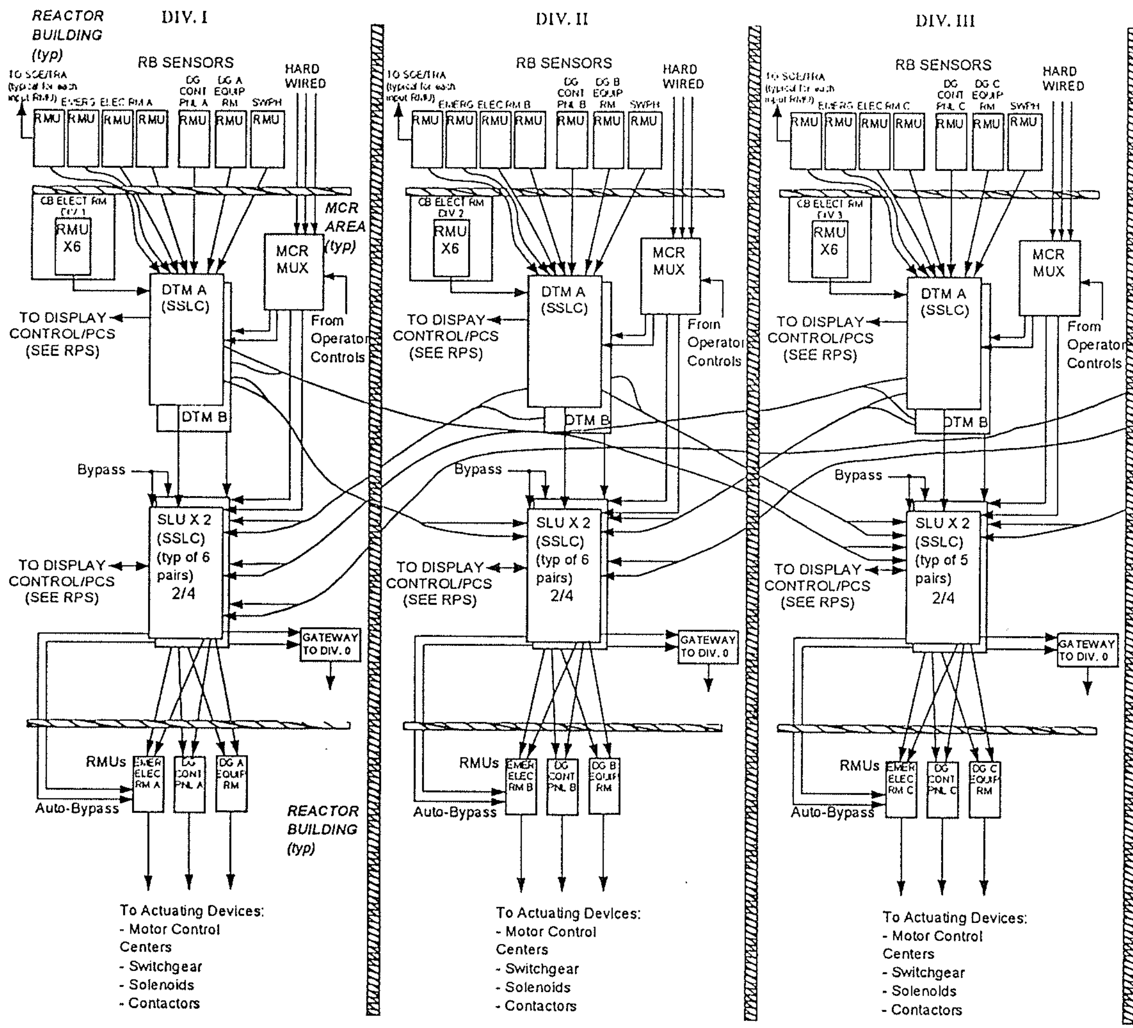


Figure 7.9-3 Configuration of RPS/MSIV and EMS Interface



NOTES:
1. Datalinks from RMUs to DTMs are dual (one link to DTM A, one link to DTM B).
2. All multiplexing links and Interdivisional links are fiber optic.
3. Data Communications Protocols
   MUX - point-to-point, deterministic
   Interdivisional data links - optical RS485
   SSLC to PCS - optical RS485
   DTM to SLU - optical or copper
   Bypass and Auto-bypass - hardwired
4. Acronyms
   CB = Control Building
   DG = Emergency Diesel Generator
   DTM = Digital Trip Module
   MCR MUX = Main Control Room Multiplexing Unit
   PCS = Plant Computer System
   RB = Reactor Building
   RMU = Remote Multiplexing Unit
   SOE/TRA = Sequence of Events/Transient Analysis
   SLU = Safety System Logic Unit
   SWPH = Service Water Pumphouse

Figure 7.9-3a Configuration of ESF and EMS Interface

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-010

PSAR Sections: Ch 7

Question Date: January 21, 1998

PSAR Question:

USNRC issued the following six Regulatory Guides in September 1997 (Lungmen PSAR was submitted in October 1997) :

1. RG 1.168 : Verification, Validation, Reviews, and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants.

2. RG 1.169 : Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants.

3. RG 1.170 : Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants.

4. RG 1.171 : Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants.

5. RG 1.172 : Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants.

6. RG1.173 : Developing Software life Cycle Processes for Digital Computer Used in Safety Systems of Nuclear Power Plants.

to replace the original draft Reg Guides DG-1054, DG-1055, DG-1056, DG-1057, DG-1058 and DG-1059 and the contents are almost unchanged. But in the various sections of Chapter 7 only commitments were given to follow the draft Reg Guides.

By comparing the dates of the issuance of those Reg Guides and the PSAR submittal, it is more appropriate to quote and commit to the officially issued Reg Guide edition.

PSAR Response:

GE agrees that the officially issued Reg Guides should replace the draft Reg Guides.

In Section 7.1.2.11.14, Reg Guide 1.168 will replace DG-1054 (same title).
In Section 7.1.2.11.15, Reg Guide 1.169 will replace DG-1055 (same title).
In Section 7.1.2.11.16, Reg Guide 1.170 will replace DG-1056 (same title).
In Section 7.1.2.11.17, Reg Guide 1.171 will replace DG-1057 (same title).
In Section 7.1.2.11.18, Reg Guide 1.172 will replace DG-1058 (same title).
In Section 7.1.2.11.19, Reg Guide 1.173 will replace DG-1059 (same title).

In Table 7.1-2, in the Applicable Criteria columns, the issued Reg Guides will also replace their draft counterparts, as above.

Similarly, wherever else one of the above six draft Reg Guides appears in the PSAR, it will be replaced with the appropriate issued Reg Guide.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-011

PSAR Sections: Sec. 7.8

Question Date: January 8, 1998

PSAR Question:

Section II. Q of USNRC memo SECY 93-087 (dated July 15, 1993) is titled Defense Against Common-Mode Failures in Digital Instrumentation and Control System" which expresses USNRC stance on design diversity requirements and it specifies :

- that the license applicant should evaluate the diversity and defense in depth design of the I/C system against common-mode failures;
- Best Estimate Method should be employed during evaluation of the various scenarios of safety analysis to verify that safety goals can still be achieved by the design diversity during common-mode failures.

Lungmen PSAR has not provided relevant descriptions with regard to the above requirements. Please provide such information or explanations.

PSAR Response:

Conformance to SECY 93-087 is discussed in PSAR Section 7.8.2.1. The implementation of diverse features to meet NRC requirements is described in Section 7.8.1.2. The evaluation required by the SECY memo to establish the diverse features was performed for the U.S. certified ABWR design by Lawrence Livermore National Laboratory (LLNL) under contract to the NRC. The results are documented in LLNL report UCRL-ID-114000, a copy of which will be furnished to ROC-AEC. Please note that a detailed chronological history of the development of this report, its conclusions, and further analyses by GE and LLNL is contained in the ABWR SSAR (Chapter 7, Appendix 7, Section 7C-4). This material was not added to the Lungmen

PSAR since it was generated for historical purposes to document the steps leading to an approved U.S. design. The LLNL methodology was incorporated into NUREG/CR-6303 (1994), which will be used for future diversity and defense-in-depth analyses. The ABWR Final Safety Evaluation Report (FSER) (NUREG-1503) also has, in Chapter 7, a detailed discussion of the diversity study and its conclusions.

While GE believes that, based on the LLNL study, the diversity features described in PSAR Section 7.8.1.2 are correct for the Lungmen project, for the FSAR GE will perform a supplementary study that will include the Lungmen-unique features of the safety systems (e.g., for RPS, seismic trip, SPTM trip, trip inhibit from turbine bypass valve opening, etc.) in conjunction with the actual safety system architecture (Safety System Logic and Control and the Essential Multiplexing System).

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-012

PSAR Sections: General

Question Date: March 6, 1998

PSAR Question:

Appendix 7.0-A; Section c.3.3 of the Standard Review Plan (SRP) has the provision that I&C safety systems incorporating digital computer technology in the reactor protection system or ESFAS comply with the NRC position on Defense-in-Depth & Diversity (D-in-D&D) described in the Staff Requirements Memorandum on SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs." BTP HICB-19 describes in detail the regulatory bases, material to be reviewed, acceptance criteria, and review process. Please provide a copy of the D-in-D&D analysis for the RPS, SSLC, and ESFAS.

PSAR Response:

Conformance to SECY 93-087 is discussed in PSAR Section 7.8.2.1. The implementation of diverse features to meet NRC requirements is described in Section 7.8.1.2. The evaluation required by the SECY memo to establish the diverse features was performed for the U.S. certified ABWR design by Lawrence Livermore National Laboratory (LLNL) under contract to the NRC. The results are documented in LLNL report UCRL-ID-114000, a copy of which will be furnished to ROC-AEC. Please note that a detailed chronological history of the development of this report, its conclusions, and further analyses by GE and LLNL is contained in the ABWR SSAR (Chapter 7, Appendix 7, Section 7C-4). This material was not added to the Lungmen PSAR since it was generated for historical purposes to document the steps leading to an approved U.S. design. The LLNL methodology was incorporated into NUREG/CR-6303 (1994), which will be used for future diversity and defense-in-depth analyses. The ABWR Final Safety Evaluation Report (FSER) (NUREG-1503) also has, in Chapter 7, a detailed discussion of the diversity study and its conclusions.

While GE believes that, based on the LLNL study, the diversity features described in PSAR Section 7.8.1.2 are correct for the Lungmen project, for the FSAR GE will perform a supplementary study that will include the Lungmen-unique features of the safety systems (e.g., for RPS, seismic trip, SPTM trip, trip inhibit from turbine bypass valve opening, etc.) in conjunction with the actual safety system architecture (Safety System Logic and Control and the Essential Multiplexing System).

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-013

PSAR Sections: General

Question Date: arch 6, 1998

PSAR Question:

The SRP (Appendix 7.0-A, Section 3.4) calls for a development process to be specified and documented such that implementation of the process gives a high degree of confidence that the functional requirements will be or are implemented in the computer system. It also requires the life cycle process plan describe a coordinated engineering process in which design outputs at each planned stage of the design process are verified to implement the input requirements of the stage. Please provide a copy of documents which describe the development process and life cycle process plan for the RPS, SSLC, ESFAS, and control computer systems.

PSAR Response:

Documents that describe the development process according to a life cycle plan are as follows:

a. Man-Machine Interface System Design Implementation Plan, Dwg. No. 31113-0A10-8101
b. Software Management Plan, Dwg No. 31113-0A51-4500
c. Software Configuration Management Plan, Dwg No. 31113-0A51-4501
d. Software Verification and Validation Plan, Dwg No. 31113-0A51-4502

Copies of the above documents will be provided as requested.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-014

PSAR Sections: General

Question Date: March 6, 1998

PSAR Question:

Compliance with the SRP/IEEE 279 requires an FMEA for the RPS and ESFAS. Please provide a copy of the FMEA(s) for the RPS, SSLC, and ESFAS.

PSAR Response:

The following material is taken from Appendix 15B (FMEA) of Chapter 15 (Accident and Analysis) of the Lungmen PSAR and also Appendix A, Probabilistic Risk Assessment (PRA), of the PSAR.

Section 15B.4 of the Lungmen PSAR explains that the FMEA for the Essential Multiplexing System (EMS) is included in the PRA fault trees and includes a separate common-cause failure analysis. EMS is the data communications system for RPS and ESFAS. The FMEA referred to in Section 15B.4 is shown in Table 15B-3. This simple, top level FMEA is included as an overview of the detailed PRA analysis.

A justification for the use of the PRA in place of the FMEA follows below. While the title is EMS, the actual PRA analysis also covers all RPS and ESFAS components and functions as part of Safety System Logic and Control (SSLC):

Equivalence of FMEA to PRA for Essential Multiplexing System

Design Considerations

Within the Lungmen Probabilistic Risk Assessment (PRA), evaluation of failure modes and effects for the Essential Multiplexing System (EMS) is performed as part of the overall safety-related instrumentation system. The fault tree analysis used in the PRA has the advantage over a conventional failure modes and effects analysis (FMEA) of inherently considering the effects of multiple failures. The design features of the EMS that are of most importance to and form the basis for this fault tree analysis are the following:

a. There is complete separation of remote multiplexing units (RMUs), transmission networks, SSLC components (DTMs, SLUs, TLUs), sensors and ECCS actuators, etc., between the four safety divisions of control and instrumentation.

b. Within a given division, the only restriction regarding assignments of sensors and actuators to RMUs is that wide-range and narrow-range reactor water level sensors cannot be input to and processed by the same RMU.

c. There is separation of DTM and TLU modules within a division along the lines of "deenergize to operate" and "energize to operate" functions, i.e., RPS, and MSIV signals are processed by different DTM and TLU modules than the DTM and SLU modules used for ECCS control and PCV isolation (PCV isolation is also deenergize-to-operate).

d. The RMUs are connected by a separate fiber optic transmission network in each division, which is a redundant or reconfigurable control data network of high reliability (MTBF=100,000 hours).

e. All data communications to and from other divisions of control and instrumentation, and all data communications to nondivisional systems are electrically isolated.

f. Comparison of a sensed input to a setpoint for generating a trip is done by a DTM. Coincident 2/4 trip logic processing for generating a divisional output trip is done by a TLU or SLU.

g. Loss of data communications in any division to the RPS (and deenergize-to-operate isolation functions) will result in a trip (and isolation, respectively) in the failed division due to the fail-safe design.

h. Manual scram is implemented by hard wire to the scram pilot valve solenoids and does not depend on the correct operation of the DTM or TLU.

i. A bypass of the RPS output logic unit is a manual division out-of-service bypass, which allows repair of the DTM or TLU of that division without a half scram condition or half MSIV isolation condition. Only one division can be bypassed at a time.

j. To reduce the probability of spurious initiation of ECCS, two SLUs are used in parallel within a division, with 2/2 voting at the final channel output to initiate equipment actuation. If one ECCS SLU is in a failed condition, it is automatically bypassed, the control room is alerted, and the remaining SLU operates with 1/1 logic until the failed SLU is restored.

k. RMUs are self-tested every 15 minutes and repaired/replaced in an average time of 4 hours.

l. Control room indications, annunciations, and alarms associated with EMS-transmitted control signals are dependent on correct operation of EMS.

m. Vital plant parameters are hard-wired to the remote shutdown panel independent of EMS.

In addition to the design features listed above, the following assumptions and groundrules also supply the basis for this analysis:

- Common-cause failure of all RMUs or all EMS networks cannot be ruled-out as impossible or incredible. The reason for this is that several potential common causes, including software design errors, can be postulated.

- The probability of common-cause failure of interdivisional EMS is extremely low. The reasons for this are the common-cause defenses built into the design-physical separation, electrical separation, asynchronous operation, optical isolation, natural convection cooling ability, and the self-testing feature.

- RMUs may be postulated to have common-cause failures of the energize-to-trip mode or the deenergize-to-trip mode, but not of both modes simultaneously.

- EMS transmission may be postulated to have common-cause failures of the energize-to-trip mode only. Failure of the deenergize-to-trip mode is considered to not be possible.

- Simultaneous failure of all RMUs or transmission networks in the energize-to-trip mode would result in an automatic scram and MSIV and PCV isolation valve closure, and loss of automatic ECCS initiation capability. Some ECCS could be initiated manually from the remote shutdown panel.

- In addition to complete failure of energize-to-trip or deenergize-to-trip functions, the RMUs may have common-cause calibration errors.

Evaluations

Both random failure and common-cause failure modes of the instrumentation system were considered in the system unavailability calculations for the PRA. Portions of the PRA taken together comprise a complete analysis of EMS failure probabilities and are provided in this section as follows:

- ***Fault Trees***. Each fault tree represents the overall complex of instrument channels, logic functions, and EMS or other transmission networks involved in generating either a reactor pressure, reactor level, or drywell pressure signal used to cause a reactor trip via RPS or to initiate the various ECCS Systems in the event of an emergency. Fault trees were developed for each signal in each electrical division. These trees are linked to the various other safety system fault trees. Events common to a number of trees are designated with identical acronyms to insure proper common cause failure treatment when these instrumentation trees are linked to the system fault trees. EMS components in the trees are represented as failure probabilities of remote multiplexing units (RMUs) and divisional fiber-optic transmission networks. Common-cause failure probabilities of these components have also been included. The instrumentation fault trees are presented in Table AA.2-23.

- *Failure Rate Data.* Failure rate data for calculating EMS component and system unavailability are provided in Table AA.2-10 for the instrumentation system components. These data include RMU and transmission network failure rates.

- *References.* Component and system unavailabilities were calculated from industry-standard failure rate data tables and from documented GE experience with similar equipment. References indicating the source of failure rate data are found in Section AA.2

Additional Considerations for Shutdown Risk

For consideration of shutdown risk, some basic features are contained in the basic design of the instrument systems and in the type and number of parameters monitored.

During shutdown, the main concern from a risk perspective is removal of decay heat from the fuel in the RPV. The large volume of water in the spent fuel pool and low probability of draining makes the risk associated with fuel pool operation relatively low. The smaller reactor pressure vessel (RPV) volume and relatively high decay heat load of the fuel increases the cooling requirements and decreases the available time to recover from loss of decay heat removal (DHR). Thus, to minimize shutdown risk, the instrumentation system must monitor RPV level and water temperature, status of makeup sources and heat sinks, and display these to the plant operators in a reliable and easy to understand manner.

Specific instrumentation features important to shutdown operations include:

- Automatic initiation of ECCS to ensure adequate RPV makeup.

- Four channels of instrumentation to allow for bypass during maintenance and testing while still retaining redundancy. (The two-out-of-four logic reverts to two-out-of-three during maintenance bypass).

- Continuous monitoring for detection of fires or flooding in safety-related and other areas.

- Operability of the RPS during shutdown to mitigate potential reactivity excursions.

- Interlocked refueling bridge operation to prevent reactivity excursion.

- Automatic isolation of shutdown cooling (SDC) on low level in the reactor pressure vessel (RPV) to ensure against fuel uncovery.

- Interlocked residual heat removal (RHR) valves (SDC and suppression pool) to reduce the potential for diversion of coolant from the RPV to the suppression pool.

- Ability to monitor radiation levels throughout the plant to detect breaches in radiological barriers.

The evaluations for shutdown risk are similar to those shown previously, but certain specific shutdown parameters are assumed to be monitored as follows:

- RPV level, water temperature, and pressure
- Neutron flux
- Drywell and wetwell pressure and temperature
- Suppression pool temperature and level
- Reactor, turbine and control building flooding level
- RHR flow rate, temperature, pump motor trip, and loop logic power failure
- RWCU outlet temperature
- Fire detection in various buildings
- Electric power distribution system parameters (e.g., power, voltage, current, frequency)
- Operation of fire water system

Evaluations for Shutdown Risk

Fault trees for instrumentation system shutdown risk evaluation, including effects of EMS failure, are discussed in Section A1.5.

Results of Analyses

The Lungmen PRA indicates that the total core damage frequency for the Lungmen ABWR design is very low. However, the results of these analyses show that, while the multi-division, bypassable, instrumentation design supports, for single random failures, the ABWR design overall core damage frequency goals, certain common-cause failures (for example, the simultaneous failure of all four divisions of EMS to transmit correct trip data) tend to dominate the event trees and override the redundant defenses built into the system configuration. The expected frequency of occurrence of common-cause EMS failures during normal operation is a function of EMS reliability, including DC power reliability. Fast recovery time from a failure due to the EMS self-test feature is possible if the common-cause failures do not occur simultaneously, but will not occur if two divisions fail simultaneously, since a plant trip occurs immediately. The probability and expected frequency of occurrence of such events in conjunction with an initiating accident or transient is extremely low. Administrative controls will be imposed to minimize the probability of progressive common-cause failures.

To further address these concerns, a separate defense-in-depth and diversity study was performed on EMS (see PSAR Section 7.8). Available safe shutdown features, including timely operator response, external to EMS and SSLC were evaluated for their usefulness in mitigating the consequences of common-cause failures. The study concluded that a diverse implementation of one division of manual HPCF initiation would eliminate potential core damage under worst-case (beyond design basis) conditions.

With the effects of common-cause failure mitigated, the EMS and interfacing instrumentation systems are seen to be single failure proof, rapidly repairable, and readily testable over the expected plant lifetime.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-015

PSAR Sections: 7.1

Question Date: March 6, 1998

PSAR Question:

1. Table 7.1-2 of the PSAR provides commitments to the acceptance criteria and guidance prescribed by the SRP with the following exceptions:

   a. R. G. 1.151 for all systems except the main steam
   b. R. G. 1.153 for all systems except the emergency chilled water systems
   c. HICB 14, 17, 18, 19, 21 for the flammability control system
   d. R. G. 1.75, 1.105, 1.152, and 1.153; and HICB 11, 14, 17, 18, 19, and 21 for the main steam system
   e. GDC 20 - 24 for emergency diesel support systems
   f. GDC 13 and 19 and R.G. 1.153 for diverse I&C
   g. GDC 24; R. G. 1.47, 1.53, 1.75, 1.105, 1.118, 1.151, and 1.153; and HICB 11, 12, 14, 17, 18, and 21 for post accident monitoring system
   h. R. G. 1.105, 1.118, 1.151, and 1.152; and HCIB 11, 12, 14, 17, 18, and 21 for remote shutdown system
   i. HCIB 17, 18, and 21 for the standby liquid control system

Please explain the reasons for these exceptions and what alternate means are planned to address these areas in the design.

PSAR Response:

1. Wherever possible, Table 7.1-2 of the PSAR is patterned after the similar Table 7.1-2 of the ABWR-SSAR and is based on applicability as documented in SRP Table 7-1. This was intended to enhance the licensing process because the ABWR-SSAR has already been reviewed and approved

by the USNRC. If an "x" is not shown on the table for a given system, it means the particular criteria is not applicable to that system, as agreed by the NRC. It is recognized that the new SRP added criteria in the PSAR table (primarily those involving digital computer control technology), which were not contained in the SSAR Table 7.1-2. However, much of this criteria was addressed in Appendix 7A of the ABWR-SSAR. Each of the noted "exceptions" are addressed explicitly as follows:

a. Per SRP Table 7-1, Regulatory Guide 1.151 applies only to systems described in PSAR Sections 7.7. Within Section 7.7, the only safety related instrument sensing lines reside in the Main Steam System. Other systems, which utilize signals from instruments on these lines, get these signals from the Main Steam System as interfaces. Hence, this guide applies only to the Main Steam System.

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

b. Reg Guide 1.153 will be added to the matrix table 7.1-2 for the Emergency Chilled Water System and will be included in the list in section 7.3.2.9.2(3).

c. The Flammability Control System (FCS) does not utilize any digital computer-based instrumentation and control. Therefore, these BTPs are not applicable to the FCS.

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

d. The listed criteria apply only to safety-related systems. Portions of the Main Steam System are safety related (Note the Automatic Depressurization System (ADS) is addressed separately as part of ECCS in PSAR Sections 7.3.1.1.1.2 and 7.3.2.1). GE agrees to add the listed criteria for those safety related portions of the Main Steam System (MS).

e. The applicability designations for Table 7.1-2 of the PSAR is patterned after Table 7.1-2 of the ABWR-SSAR if the system title and criteria are the same in both designs. GDCs 20-24 are not considered applicable to the EDG support systems of themselves (as agreed by the NRC) because they are at the individual subsystems to each of the three EDGs. Rather, these criteria apply to the nuclear protection systems at the system level. In that context, each EDG, in association with its support systems, certainly meets the requirements of these criteria (i.e., automatic initiation, reliability, testability, independence, etc.)

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

f. GDCs 13, 19 and R.G. 1.153 were not applied to Section 7.8 per the Lungmen Standard Review Plan (SRP) Table 7-1. PSAR Table 7.1-2 is based on this SRP, hence, these criteria are not applied to Diverse I&C.

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

g. Consistent with the ABWR Certified Design, the Lungmen NPS post accident monitoring (PAM) is not a designated "system" of itself. Rather, PAM is accomplished by assimilation of a large set of discrete data, as defined by Regulatory Guide 1.97, which is gathered from the variables and parameters of many existing systems. Sections 7.5.1.1 and 7.5.2.1 of the PSAR provide an assessment of Regulatory Guide 1.97, and identifies how the variables available from existing systems fulfill the PAM requirements. Therefore, Regulatory Guide 1.97 is addressed in detail for the PAM. However, the criteria identified in this question are addressed by the existing systems in accordance with the applicability required in Table 7-1 of the SRP.

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

h. HICB 11 provides guidelines for reviewing the use of isolation devices in instrument and control systems, including isolation devices that provide isolation between redundant safety divisions. RSD utilizes such isolation devices to maintain isolation between redundant safety divisions. Therefore, this Branch Technical position does apply to RSD, and PSAR Table 7.1-2 will be updated accordingly.

HICB 12 provides guidelines for reviewing the process an applicant/licensee follows to establish and maintain instrument setpoints. The instruments used by RSD belong to the systems that interface with

RSD. Therefore, this Branch Technical Position does not apply to RSD. No changes will be made to the PSAR as a result of the response to this question item.

HICB 14 provides guidelines for evaluating software life-cycle processes for digital computer-based instrumentation and control systems. RSD is not a digital computer-based I&C system. Therefore, this Branch Technical Position does not apply to RSD. No changes will be made to the PSAR as a result of the response to this question item.

HICB 17 provides guidelines on self-test and surveillance test provisions that apply to digital computer-based I&C systems. RSD is not a digital computer based I&C system. Therefore, this Branch Technical Position does not apply to RSD. No changes will be made to the PSAR as a result of the response to this question item.

HICB 18 provides guidelines for reviewing the use of PLCs in I&C systems. RSD does not use PLCs. Therefore, this Branch Technical Position does not apply to RSD. No changes will be made to the PSAR as a result of the response to this question item.

HICB 21 provides guidelines for reviewing digital system real-time performance and system architectures in I&C systems. RSD is not a digital computer-based I&C system. Therefore, this Branch Technical Position does not apply to RSD. No changes will be made to the PSAR as a result of the response to this question item.

R.G. 1.105 describes criteria for instrument setpoints for safety-related systems. The instruments used by RSD belong to the systems that interface with RSD. Therefore, this U.S. NRC Regulatory Guide does not apply to the RSD. No changes will be made to the PSAR as a result of the response to this question item.

R.G. 1.118 describes criteria for the periodic testing of electric power and protection systems. RSD is classified as one of the Safe Shutdown Systems. RSD is not classified as an Electric Power System or one of the Protection Systems. Therefore, this U.S. NRC Regulatory Guide does not apply to RSD. No changes will be made to the PSAR as a result of the response to this question item.

R.G. 1.151 describes criteria for instrument sensing lines. The instruments (including sensing lines) used by RSD belong to the systems that interface with RSD. Therefore, this U.S. NRC Regulatory Guide does not apply to RSD. No changes will be made to the PSAR as a result of the response to this question item.

R.G. 1.152 describes criteria for digital computers in safety systems of nuclear power plants. RSD is not a digital computer-based I&C system. Therefore, this U.S. NRC Regulatory Guide does not apply to RSD. No changes will be made to the PSAR as a result of the response to this question item.

i. HCIB 17, 18, and 21 apply to digital computer-based I&C systems. The Standby Liquid Control System (SLC) is not a digital computer-based (software) system at the PSAR design stage; therefore, these criteria do not apply at this time. However, the SLC could ultimately be either a software based system or a hardware based system, depending on the bid result. If the SLC changes to a software system, these criteria will be added for the FSAR.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-016

PSAR Sections: 7.2

Question Date: March 6, 1998

PSAR Question:

Section 7.2.1.1.4.2 of PSAR; please explain why the RPS does not have a "High Main Steamline Radiation" trip?

PSAR Response:

The elimination of the automatic reactor shutdown feature associated with the Main Steam Line Radiation Monitor (MSLRM) is based on the GE safety analysis contained 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". NEDO-31400A. which evaluated and justified the removal of the above MSL high radiation trips, was reviewed and accepted by the US NRC. Many GE BWRs have now removed the reactor scram and vessel isolation trip functions that were previously associated with the MSLRM as a result of the evaluation.

The elimination of the automatic reactor scram function of the MSLRM will reduce the potential for unnecessary reactor shutdowns caused by the spurious actuation of the MSLRM trips and will increase the plant operational flexibility. This change also reduces the number of the RPS trips, thereby enhancing RPS optimization.

Additional information on the reason and background of the MSLRM change can be found in NEDO-31400A.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-017

PSAR Sections: 7.4

Question Date: February 27, 1998

PSAR Question:

Section 7.4.2.2 of PSAR; does the remote shutdown station utilize the EMS or NEMS? Does it use any computer or microprocessor-based components? If so, please describe those and their role in supporting remote shutdown operations.

PSAR Response:

Signals for the remote shutdown panel alarms discussed in PSAR Section 7.4.1.2.4 (1) are sent to the main control room via the EMS. The remote shutdown panels do not utilize the NEMS. There will not be a remote shutdown panel computer system programmed to perform all remote shutdown system functions. However, given the current state of technology, some microprocessor-based components may be mounted on the remote shutdown panels (i.e. process signal indicators, M/A stations, analog signal square root devices, analog signal averaging devices, etc.)

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

ROCAEC Review Comments:

1. GE's commitment for the shutdown panels is analog devices with the relevant codes and standards (C&S). If digital devices are used, C&S should be using the ones that govern digital devices.
2. If there are digital equipment on remote shutdown panel in the future then it will violate 7.8.1(4) (page 7.8-1). This is a fundamental question, which needs further clarification and a position be proposed.

Further Clarifications:

1. To the extent that digital devices might be used on the RSD panels, Codes & Standards applicable to the types of processors contained within these devices would also be followed. At this time, GE has not received any RSP panel vendor submittals. Therefore, it is not yet known what types of devices will be proposed for the RSD panels. GE will insure that the safety function of RSD panels will not be lost because of possible common mode failure of digital devices that might be used on the RSD panels.

2. Use of simple, dedicated and diverse software based digital equipment on the remote shutdown panels is allowed by Branch Technical Position HICB-19, "Guidance for Evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems." To clarify the discrepancy noted in the review comment, PSAR paragraph 7.8.1(4) on page 7.8.1 will be revised to read as follows:

" (4) Long term shutdown capability provided in a conventionally hardwired remote shutdown system with 2 divisional panels containing analog or simple, dedicated and diverse software based digital equipment. Local displays of process variables in RSD system are continuously powered and so are available for monitoring at any time."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-018

PSAR Sections: 7.2.1

Question Date: February 20, 1998

PSAR Question:

Section 7.2.1.1.4.2 described the signal transmission method of various trip signals of RPS (see second paragraph of page 7.2-5) in that all signals use EMS for transmission except NMS, MSIV closure and TB-trips. And TB-trips use hardwired transmission method. This statement is inconsistent with the explanation for each individual signal in the following two places :

1. Item (2), (d) (page 7.2-6) stated that MSIV position switches signal transmission uses EMS.

2. Item (4), (e) (page 7.2-8) stated that seismic activity acceleration switches also use hardwired connection.

Please clarify.

Response:

1. The next revision of the PSAR will change section 7.2.1.1.4.2 item (2) (d) to state that the MSIV position switches are hardwired to their associated SSLC DTMs.
2. The next revision of the PSAR will change section 7.2.1.1.4.2 item (4) (f) to state that the seismic activity acceleration switches are transmitted to their associated SSLC DTMs via the EMS.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-019

PSAR Sections: 7.2.2

Question Date: February 20, 1998

PSAR Question:

Section 7.2.2.2.3.1, item (10), (b) (in page 7.2-38) described the functional tests of air header dump valve. Please explain where in Chapter 16 (Operation Procedures) this requirement of functional test is mentioned?

Response:

Information pertaining the air header dump valve can be found in Section 16B.3.3.1.1 (page 16B.3.3-6).

Subsection 7.2.2.2.3.1, item (10) (c) of Lungmen PSAR addresses the requirement of testing of RPS backup scram valves during each refueling outage.

This requirement will be carried out in accordance with SR 3.3.1.2.5 under Section 16.3.3.1.2, Reactor Protection System (RPS) and Main Steam Isolation Valve (MSIV) Actuation of Lungmen PSAR. The 18 month frequency specified in SR 3.3.1.2.5 is based on the ABWR expected refueling interval and the need to perform this surveillance under the condition that apply during a plant outage to reduce the potential for an unplanned transient.

Details of SR 3.3.1.2.5 can be found on page 16B.3.3-80 under Section 16B.3.3.1.2 of Lungmen PSAR

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: 07-020

PSAR Sections: 7.2.2

Question Date: February 20, 1998

PSAR Question:

Section 7.2.2.2.3.1 items (8) and (12) are contradictory. Please clarify : is the bypass signal of the reactor trip signal from the fast closure of T/B stop valve and control valve related to the first stage pressure of the T/B or to the reactor mode switch position ? Please also modify the inconsistencies noted.

Response:

The TSV closure and the TCV fast closure reactor scram is automatically bypassed when :

- APRM simulated thermal power of the NMS is below 40% of the rated reactor power output. Please note that
- Sufficient number of the Turbine Bypass Valves (TBV) are opening as indicated by their 10% position sensors.

The next revision of PSAR will change section 7.2.2.2.3.1 (8) to :

- Replace the turbine first stage pressure with the APRM simulated thermal power as the appropriate variable providing the desired measurement of power level .
- Describe the TBV position monitored by RPS as a permissive to bypass the TSV closure and the TCV fast closure reactor scram.

The logic for the automatic bypass of reactor scram on the TSV closure and the TCV fast closure does not receive permissive signals from the reactor mode switch. The next revision of PSAR will change section 7.2.2.2.3.1 (12) to clarify.

The description contained in PSAR Section 7.2.2.3 (4), "Plant Load Rejection", and Section 7.2.2.3 (4), "Turbine Trip" provides additional useful information.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-021

PSAR Sections: 7.2, 7.3

Question Date: February 26, 1998

PSAR Question:

Regulatory Basis: GDC 23 - Protection System Failure Modes. Please provide detailed information about Failure Modes and Effect Analysis of SSLC design, specifically, please emphasize on "wrong output" failure mode and subsequent effect of SSLC.

Response:

- Section 15B.4 of the Lungmen PSAR explains that the Failure Mode and Effect Analysis (FMEA) for the Essential Multiplexing System (EMS) is included in the PRA fault trees and includes a separate common-cause failure analysis. The FMEA referred to in section 15B.4 is shown in Table 15B-3. This simple top level FMEA is included as an overview of the detailed PRA analysis. The actual PRA analysis also covers RPS and ESFAS components and functions as part of the Safety System Logic and Control (SSLC).
- For the effect of "Wrong Output" failure on SSLC, please note that PSAR Sections 7.1, 7.2, and 7.3 establish that no single failure, anywhere in the system, can affect more than one division. Since the SSLC implements two-out-of-four voting logic, a single failure (e.g., wrong output) may cause the logic revert to two-out-of-three voting.

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: 07-022

PSAR Sections: 7.2, 7.3

Question Date: February 26, 1998

PSAR Question:

Regulatory Basis: BTP 14 Guidance on Software Reviews for Digital Systems. Please explain how SSLC software specification and design will satisfy consistency and completeness criteria as required by BTP 14? Please provide technical rationales of how these criteria can be determined?

Response:

The SSLC software will be designed in according to the SSLC Hardware/ Software Specifications (HSS). The SSLC HSS will specify that the software be designed to comply with BTP 14.

PSAR Section 7.8.1.3, "Common Mode Failure Defenses Within Safety System Design", and Section 7.1.1.2.1.3(2), "Software Development" contain the software design requirements.

The Lungmen DCIS Software Verification and Validation Plan (31113-0A51-4502) describes the software V&V activities to be performed during the development of software-based products. For safety-related (Class S) software. Section 5.2.1 of the SVVP describes an Independent V&V Plan (IVVP) to be developed by the Lungmen Independent Review Team (IRT) in accordance with the requirements defined in Reg Guide 1.168. The IVVP describes the V&V activities performed by the Lungmen Independent V&V Team (IVVT) for Lungmen DCIS safety-related software.

Appendix C of the SVVP describes the generic independent V&V tasks to be performed by the IVVT. The tasks include assessments of engineering documents for traceability, correctness, completeness, accuracy, and consistency. This set of generic V&V activities will be applied by the IVVT, as appropriate, to the DCIS software engineering process defined in the Software Management Plan (SMP). Appendix C defines, for each life-cycle phase, the inputs required to perform the task description and the resulting outputs. For example, in the Design Phase, for the Design Traceability Analysis task (Item 3 of Appendix C), tracing the software requirements specification to the software design specification (and vice versa), the SRS, SDD, interface design documentation, and interface requirements documentation will be used to determine that relationships are correct, accurate, etc. (including consistent and complete). The documents are checked against each other for consistency (i.e., no contradictions) and completeness (i.e.,-full implementation of the requirements) per BTP-14. The findings will be documented in a task report and an anomaly report.

The IVVT will utilize the methods or combinations of methods presented in GENE Engineering Operating Procedures, IEEE-1012, and IEEE 1028, depending on the necessity of the specific V&V task or requirements criteria. The IVVP will fully describe the review and documentation methods employed in the generation of the independent V&V outputs.

The following Table 1 indicates how a determination of completeness and consistency (reflecting the definitions in BTP-14) is reached at each phase of software development.

Table 1 - Determination of Software Consistency and Completeness

Software Phase	Representation	Completeness	Consistency
Software Requirements Spec.	Information flow and interfaces, definition of algorithms and equations, detailed control logic and data operation	Traceability Analysis, Design Review	Traceability Analysis, Design Review
Design	Software	Logic Analysis	Logic Analysis

Software Phase	Representation	Completeness	Consistency
Specification	architectural layout, software structure, data flows, and algorithms	Review, Traceability Matrix	Review, Traceability Matrix
Coding	High-level language (not assembly language)	Code Review, Source code Traceability Matrix, Software code interface review, Source code documentation evaluation	Code Review, Source code Traceability Matrix, Software code interface review, Source code documentation evaluation
Integration Testing	Test cases	Design Review, Traceability Matrix	Design Review, Traceability Matrix

Basis: 1. Lungmen DCIS Software Management Plan (31113-0A51-4500), Section 5.4.1.1, oftware Coding is the translation of the detailed software design (as specified in the SDS) into machine-readable code using a predetermined high level programming language. The software programs must be implemented in accordance to the Software Conventions and Guidelines Documents. Each module of the source code shall contain comments of sufficient detail to allow a qualified software engineer to understand the design intent and logic flow. At a minimum, each module shall contain a statement of the procedure's purpose or function, and a brief description of the algorithm implemented. Each software module shall identify whether or not the module is part of, or contributes to a safety-related function.

For safety-related applications, the standards, conventions, and techniques used to implement the software design for the support (non-deliverable) software shall be the same as the process required for the deliverable software..."

2. Lungmen DCIS Configuration Management Plan (31113-0A51-4501), Section 3.7, "Formal configuration reviews and audits shall be performed on the software CIs [Configuration Items] to ensure the completeness of the software-based products."

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: 07-023

PSAR Sections: 7.2, 7.3

Question Date: February 26, 1998

PSAR Question:

Regulatory Basis: IEEE 603 5.14 - Human Factors Considerations. Please explain how Lungmen's human-computer interface design can prevent mode confusion (or situation awareness) problem. (Mode confusion of a computerized control system has been identified as a major cause of several airplane crash accidents in aviation industry)

Response:

The potential for adverse affects (e.g., mode confusion) on human "situational awareness" (footnote 1) have been investigated in the aviation industry following accidents involving highly automated (fly-by-wire) aircraft with "glass cockpits". Investigations and related research suggest that automation complexities, and the predominantly computer display-based cockpit controls are key factors behind pilot loss of "situation awareness" in some of the accidents. "Situational awareness" is addressed by the HFE-related program plan for Lungmen NPS [PSAR Chapter 18, Table 18.7-1, Part II(1)(b)(iii)]. Analyses are done to confirm that operators can perform their tasks while maintaining situational awareness, workload and vigilance [PSAR Chapter 18, Section 18.7.2.4 (Allocation of Functions) and Table 18.7-1, Part IV(1)(e)]. Task Analysis identifies tasks critical to safety in terms of importance for function achievement, potential for human error, and impact of task failure. Where critical functions are automated, the analyses (footnote 2) address the human tasks including the monitoring of the automated functions and the backup manual actions which may be required if an automated function fails [PSAR Chapter 18, Table 18.7-1, Part VI(1)(c)]. Situation awareness and crew errors are two of the performance measures that are the basis for evaluating test results conducted as part of the Human Factors Verification and Validation (V&V) process [PSAR Chapter 18, Table 18.7-1, Part VII(1)(f)(ii and iii)].

Key elements of the human-computer interface design (Human-System Interface or HSI as it is called in PSAR Chapter 18) include proven technologies, appropriate human factors principles, and human-centered automation. Each of these elements account for human cognitive strengths and weaknesses which are expressly included in the Lungmen NPS function allocation (automation) philosophy and criteria. The human-centered automation approach for Lungmen can be summarized as:

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

The following table identifies many features and elements of the Lungmen HSI design that preclude loss of situation awareness. These features of the Lungmen HSI design will be verified and validated using the full-scope simulator to confirm (a) that mental and physical tasks do not exceed the performance capabilities of the operators, (b) that "situational awareness" can be acquired and maintained, and (c) that potentially new types, and possibilities, of human error have not been introduced.

Equally important as HSI design to "situation awareness" (and reliable operator performance in general) are training and procedures. The following methods can be applied to promote and maintain "situation awareness":

- Educating operators and maintainers on what the automation does (both well and not well) and what the automation does not do. (This includes understanding the operator's supervisorial, or mission manager, decision-making role.)
- Following procedures expressly developed for effective transfer and communication of unit information during shift changeovers.
- Using the training simulator to test for overdependence on automation (i.e., identify conditions when operators are reluctant to act despite being certain about abnormal conditions). The objectives include encouraging operator discretion, reinforcing the operator being "in the loop", promote learning from potential mistakes, and motivating operators to learn by giving them the opportunities to experiment.
- Acquiring and maintaining familiarity with plant processes and equipment by conducting periodic plant "walkdowns" to observe equipment and work activities.

Features of the Lungmen Human-System Interface (HSI) Design That Preclude Loss of Situation Awareness

Potential Deficiencies of Automation Design and HSI Design Associated with Loss of Situation Awareness	Features of the Lungmen Human-System Interface (HSI) Design That Preclude Loss of Situation Awareness
Mode confusion	• Mode changes are designed to be deliberate actions and they are invoked using fixed-position switches [PSAR Section 18.6.1] to keep the operator "in the loop". • Alarm suppression is based on operating modes (e.g., plant modes, system modes, equipment modes) [PSAR Section 18.4.2.12] to eliminate irrelevant and ambiguous information. • Electronic display of equipment tagout status.
The operator has difficulty managing task support resources. Provisions for detecting and recovering from human error are limited. There are inadequate provisions for manual aids.	• Electronically displayed procedures (in logic or flow chart form) with the following features: – Ability to verify the operator decisions yet the operator retains ultimate authority and control – Automatic logging of operator deviations from the procedural options available to them on the displays. – Ability to retrace certain procedure steps (except operator control actions) to assure that proper status of systems or components is maintained. – Operator can access a particular control from either a system standpoint (e.g., from a P&ID-type display) or from a functional standpoint (e.g., a procedure display). – Automated tracking for Emergency Operating Procedures • Monitoring of plant Technical Specifications for violations of Limiting Conditions of Operation and presenting recovery actions to the operator. [See Response to PSAR Batch 6, Question 18-001, Item 3. Transmitted under letter GEAE-1998-0094 (20Feb98).] • The control room HSI design and control room layout accommodate operator use of hardcopy procedures, large engineering drawings, clipboards, notepads, etc.
The operator has difficulty understanding the actions and status of the automation. There is inadequate time for operator interpretation, evaluation, and response.	• The operator is provided with task-relevant information and automated actions (taken, in progress, and pending) status. • Automated unit startup and shutdown sequences include hold points (break points) that provide ample time for operator decision-making [See PSAR Chapter 7, Section 7.7.1.5.2]. Display designs are intuitive (for ease of use) and highly discriminating (by mode, by function, by system, etc.). • System operations (i.e., procedures for pre-operations, operation, shutdown, and surveillance testing) do not require complex or time-consuming "programming" (e.g., logical settings of modes, inhibits, interlocks, data entry, reading and interpreting displays).

Features of the Lungmen Human-System Interface (HSI) Design That Preclude Loss of Situation Awareness

(continued)

Potential Deficiencies of Automation Design and HSI Design Associated with Loss of Situation Awareness	Features of the Lungmen Human-System Interface (HSI) Design That Preclude Loss of Situation Awareness
Presentation of information is highly serial (sequential) making it difficult to navigate, assimilate, and share views of information (i.e., a "keyhole" or "tunnel" effect)	• The operator crew is provided with both serial data presentation (primarily at the operator Main Control Console) and parallel data presentation (primarily at the Wide Display Panel, WDP). • The spatial arrangement of the control room panels allows the entire control room operating crew to conveniently view information presented on the WDP [PSAR Chapter 18, Section 18.4.2.1(17)]. The crew size and panel arrangement are conducive to teamwork and crew interaction (joint monitoring, sharing of information, task delegation, notification of key actions taken at control panels).
The automatic controls design (by intention or arbitrarily) limit the extent of human operability and direct ("hands on") control of equipment. Operators experience complacency, lack of vigilance, boredom, etc.	• The HSI is designed for the capability to conduct all plant operations in an operator manual mode, and for operators to assume manual control by normal procedural methods and whenever operators elect to do so at their discretion [PSAR Chapter 18, Section 18.4.2.1(7)]. The operators retain ultimate authority and decision-making responsibility. • Operator preferences, experience, familiarity, and acceptance are factored into the design by having TPC operations personnel participate in the control room design development (from specification through verification).
There is insufficient feedback and warnings to effectively anticipate problems, and computer displays are the only sensory input media for the operator.	• The Wide Display Panel provides fixed-position, plant-level and system-level alarm tiles needed by the operators [PSAR Chapter 18, Section 18.4.2.11]. • Safety Parameter Display System (SPDS). The SPDS aids operators during abnormal and emergency conditions in (a) determining the unit safety status, (b) assessing whether abnormal conditions warrant corrective actions by operators to prevent core damage, (c) monitoring the impact of engineered safeguards or mitigation activities, and (d) executing symptom-based emergency operating procedures. [PSAR Chapter 18, Section 18.4.2.10]. • Information available to the operator includes diagnostic and trend monitoring data (e.g., equipment vibration monitoring), and information regarding system fault detection, identification, verification, and recovery. • The operator crew has closed circuit television (CCTV) and intraplant voice communication systems.

Footnotes:

1. Operator "situational awareness" means having enough cognition to integrate many inputs (e.g., event-oriented and symptom-oriented status, real-time data, performance information, and crew communications) into a single, coherent "picture" (understanding) of the overall situation and operational status. It is especially pertinent where the human has a role as a mission manager, operating a computerized system and primarily monitoring rather than controlling equipment in a direct, hands-on manner. Navy combat operation crews use the phrase "having (or being in control of) the bubble" to describe this state of awareness. "Losing the bubble" is a state of incomprehension or misunderstanding, even when reliable information is available.

2. The allocation of functions (i.e., extent of automation) will be verified and validated, and the Task Analysis will acknowledge that the operators have a "supervisory" (or mission manager) role. Appropriate human factors principles will be applied to reinforce the cognitive and "situation awareness" skills that the TPC operator population possesses (from experience with existing plants) rather than trying to establish an altogether new cognitive model.

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: 07-024

PSAR Sections: 7.2, 7.3

Question Date: February 26, 1998

PSAR Question:

Regulatory Basis: GDC 20 Protection System Functions. Explain why Lungmen Protection System functions do not include software failure as a trip function? (cf. The digital protection system developed by General Atomic for TRIGA Mark I reactor has a software failure trip condition.)

Response:

The description contained in PSAR Section 7.8.1.3, "Common Mode Failure Defenses Within Safety System Design", and Section 7.1.1.2.1.3(2), "Software Development" addresses this concern :

- The SSLC design is modularized for both hardware and software.
- The software design process specify the use of modular code. Code is segmented by system and functions. Program code for each safety system resides in independent modules which perform setpoint comparison, voting and interlock logic
- The system is designed such that the failure of a software module, programmed into PROM as firmware, of any protective function can only affect the logic in one division.

Common-mode failures within the software is very unlikely to happen for the following reasons:

- Simple, modular program
- A strong V&V program
- Continuous self-test

Additionally , even if a common mode failure were to occur, the operator will have enough resources available to effectively handle the situation.

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: 07-025

PSAR Sections: 7.2, 7.3

Question Date: February 26, 1998

PSAR Question:

Regulatory Basis: GDC 29. INEL research (sponsored by NRC) indicated that the introduction of computerized man-machine interface will increase the rate of operator commission error, please analyze the impact of such commission error on the probability of accomplishling safety functions as required by GDC 29.

Response:

GE does not concur with the cited INEL unqualified statement. The statement overlooks key factors which can reduce the rate of error of commission such as training, human engineering features implemented in the man-machine interface, fault tolerant software designs, diagnostics capability, diversity and defense-in-depth (e.g., the more conventional man-machine interface of the Remote Shutdown Facility), etc.

An extensive human reliability analysis has been performed for the operator actions included in the Lungmen NPS PRA, PSAR Appendix A, Attachment AH. The analysis used state-of-the art techniques but did not address error of commission at this early stage of the design. Human reliability analysis will continue during the design phase to support design decisions. Analysis of the error of commission will be performed when adequate design details are available. Attachment AH of Appendix A will be updated for the FSAR and will include assessment of the risk-significant error of commission.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-026

PSAR Sections: 7.2, 7.3

Question Date: February 26, 1998

PSAR Question:

Regulatory Basis: RG 1.70 Standard Format for PSAR. Section 7.2 of RG 1.70 requires comparision of similar protection system designs. Please provide such information for LungmenRPS (eg., compare Lungmen design with K6/K7.)

Response:

The response to the above question will focus on a comparison between the Lungmen NPS design and the US ABWR certified design since a stated high level goal of Lungmen NPS is to utilize a design that is approved in the country of origin. (Comparisons to other operating ABWRs are not pertinent as they were constructed according to regulations in another country.)

The RPS design of Lungmen NPS is identical to the that of the US ABWR certified design except for the following features:

- Reactor scram on high seismic activity
- Elimination of MSL high radiation reactor trip
- Full load rejection and turbine trip, without reactor scram capability. This requires automatic bypass of reactor scram on Turbine Stop Valve (TSV) closure and Turbine Control Valve (TCV) fast closure if sufficient number of the Turbine Bypass Valves (TBV) are opening as indicated by their 10% position sensors.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-027

PSAR Sections: 7.2, 7.3

Question Date: February 26, 1998

PSAR Question:

Regulatory Basis: IEEE 603 section 5.9 - Control of Access. Please explain how Lungmen digitial I&C system will protect itself against malicious intrusion from software standpoint of view. (eg., virus, logical time-bomb, network intrusion, etc.)

Response:

The description contained in PSAR Section 7.8.1.3, "Common Mode Failure Defenses Within Safety System Design", and Section 7.1.1.2.1.3(2), "Software Development" addresses this concern :

- Software embedded in PROMs (firmware), programs cannot be changed by operator or external signals
- Built in test and diagnostic
- Protection inherent in redundancy
- Protection inherent in plant operation and maintenance
- Encoded data format
- Limited access to logic circuits /components
- Strict configuration management practice

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-028

PSAR Sections: 7.2, 7.3

Question Date: February 26, 1998

PSAR Question:

Regulatory Basis: RG 1.53, IEEE 379, IEEE 603 Single Failure Criteria, BTP-19 - Defense-in-Depth and Diversity, etc. Current single failure/common mode failure analyses emphasize on component sharing induced failures. However, for software-controlled systems, interaction induced failure could have more impact to common mode failures. Please provide DI&DD analysis which takes interaction-induced common mode failure into consideration.

Response:

As described in PSAR Sections 7.1, "Introduction", 7.2, "Reactor Trip System - Instrumentation and controls", and 7.3, "Engineered Safety Features Systems", the concept of Defense-in-depth is maintained by:

- Control systems are independent of RPS and ESF in separate processing network.
- Fail-Safe functions (RPS, MSIV) and fail-as-is functions (ESF) are performed in separate processing channels within each division
- ESF functions are further subdivided into individual channels of high pressure (e.g., HPCF, RCIC) and low pressure (e.g., RHR, ADS) channels within each division

The description contained in PSAR Section 7.8.1.3, "Common Mode Failure Defenses Within Safety System Design", and Section 7.1.1.2.1.3(2), "Software Development" addresses this concern :

- Software permanently embedded as firmware in controller ROMs
- Modular software distributed among separate controllers
- Modules perform limited tasks under control of simple operating system

- Software is segmented by task and system
- Auto diagnostic software and hardware monitor functional software in each controller
- Comprehensive V&V program, including integrated system testing & site testing.

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: 07-029

PSAR Sections: 7.2, 7.3

Question Date: February 26, 1998

PSAR Question:

IEEE 603 Section 4.9 requires the identification of the methods used to determine the reliability of safety system.

Please explain how the reliability (2.95E-04 as shown in Appendix A page A.2-47 of Lungmen Project ABWR PRA) for digital trip module of RPS is achieved.

Response:

The value of 2.95E-4 is the probability that the module will not be available when needed. The value has been derived from the following input:

1) The module failure rate is 5.0E-6 per hour
2) The module functionality is tested automatically every 0.5 hr (conservative assumption) and also tested quarterly (2190 hrs).
3) The probability that the automatic 30-min. testing fails to detect a failure is 0.05
4) The time to repair or replace a failed module is 4 hrs.

From the above input, the module unavailability is estimated as follows:

$$P = [0.95 \times (0.5/2 + 4) + 0.05 \times (2190 / 2 + 4)] \times 5.0E\text{-}6$$

$$= 2.95E\text{-}4$$

The first term in the bracketed expression represents a failure occurring between two successive automatic tests which is successfully detected and repaired or replaced in 4 hours. The second term represents failures which were not detected by the automatic test but were detected by the quarterly test, then repaired or replaced in 4 hours.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-030

PSAR Sections: 7.2 Reactor Trip System

Question Date: February 26, 1998

PSAR Question:

Regulatory Basis: GDC 21, RG 1.118, IEEE 338 Periodic Survelliance Testing. Conventional pre-selected input-output correctness checking may not identify logical/algorithmic errors that remain embedded in software. Please explain and justify how Lungmen on-line testing approach can identify software logical errors.

Response:

The description contained in PSAR Section 7.1.1.2.1.3(2), "Software Development" addresses the concern expressed in the above question by having :

- Software is embedded in PROMs (firmware), and the programs that implement logical functions cannot be changed by operator or external signals
- A strong V&V program which includes baseline design review, hardware and software integration testing, field installation testing
- Logic changes require changing memory chips
- New firmware will be supplied fully tested and validated
- Installation and testing at site will be controlled by the V&V program and the site QA process.

There is no need, in general, to provide on-line identification of software logical errors. Adding complex. safety-related, self-test routines to the simple functional logic for RPS and ESF increases the difficulty of V&V and may lower reliability. However, some runtime monitoring is provided, as explained below.

The formal V&V program under which Lungmen safety-related software is developed certifies the correctness and reliability of the integrated software and hardware assemblies

throughout the development life cycle and during the factory acceptance test. Further testing is performed during installation and startup at the plant site. In the unlikely event that the embedded software in the protection system controllers changes during operation (due to memory chip failure or EMI, for example), the continuous on-line self-diagnostic program that checks the 'health' of the controllers' CPU, memory, and I/O is expected to detect the majority of software and hardware failures. The parameters checked by the diagnostics include the following (see PSAR Section 7.1.2.1.6, SSLC Inservice Testability, paragraph 2(a)):

a. Monitoring of overall program flow
b. Reasonableness of process variables
c. RAM and PROM condition
d. Device interlock logic
e. Hardware and software watchdog timers
f. Continuous error checking of all transmitted and received data on the serial data links of each SSLC controller by parity check, checksum, and/or CRC techniques
g. Ringback of contact closure outputs

These features meet the NRC guidelines given in Branch Technical Position HICB-17 (Guidance on Self-Test and Surveillance Test Provisions)

The functional safety logic is checked off-line during periodic surveillance testing as specified in the plant Technical Specifications. A surveillance test controller (STC) is provided in each safety division to inject test patterns (normal and abnormal) that exercise the functional logic. The STC then monitors the effect on the system outputs. The on-line diagnostics and off-line surveillance testing is expected to provide full coverage for detection of errors. Note that the ESF logic trains are checked for logical errors and other faults by means of the dual processing of the control programs in the two SLUs and dual, 2-out-of-2, RMU outputs in each train. Both trains must agree for a control output to occur. RPS outputs are protected from single division software errors by the 2-out-of-4 configuration of the scram pilot valve load drivers. Both RPS and ESF input signal processing is protected by the 4-division redundant sensor arrangement and the 2-out-of-4 trip voting. On a single-failure basis, this provides more than adequate protection for both failure-to-trip and inadvertent trip conditions.

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: 07-031

PSAR Sections: 7.9 Data Communications Systems

Question Date: February 26, 1998

PSAR Question:

EMX provide information to both safety systems (e.g., RPS, ESF, etc.) and non-safety systems (e.g. process computer, alram, etc.) Please explain the detail design of EMX especially those parts for assuring the interaction between EMS and non safety systems will not violate "Independence" requirement.

Response:

The detailed design of the EMS for the interaction between EMS and non-safety systems will include the following design features to assure independence of safety systems;

- Class 1E data gateways to carry 1E signals from EMS to NEMS. These gateways will allow transmission of data only in one direction from EMS to NEMS and also the physical independence between the EMS and NEMS is achieved by the use of fiber optic communication.
- Dedicated fiber optic datalinks between Safety Systems and non-Safety Systems. Again, the independence between the EMS and NEMS is achieved by the use of fiber optic communication.
- Software failures on the non-safety system side will not propagate back to the safety system or EMS.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-032

PSAR Sections: sec 7.8

Question Date: February 26, 1998

PSAR Question:

PSAR Figure 7.8-4 Implementation of Additional Diversity in SSLC to Mitigate Effects of Common-Mode Failures showed the diversity in SSLC to mitigate effects of common-mode failures, HPCF C manual initiation, manual scram and manual isolation.

1. the figure indicated that :

   a. pressure sensor PT025 sends pressure signal simultaneously to PI625 and RMU.

   b. Water level sensor LT003 sends water level signal simultaneously to LI603 and RMU.

   c. Flow rate sensor FT009 sends flow rate signal simultaneously to FI609 and RMU.

   If the above sensors (PT025, LT003 or FT009) failed, it will cause the diverse display function to fail too. Please clarify or modify the system.

2. When abnormal conditions occurred to pressure, water level or flow rate, is the pressure display PI625, water level display LI603 or flow rate display FI609 accompanied by sound alarm? If not, how does the operator know the emergency state has been reached if he/she did not see the abnormal display?

3. Please describe in detail and prove that HPCF C manual initiation and scram, RWCU and RCIC manual isolation are completely independent from SSLC and will function properly when SSLC failed. Also, please explain whether the above equipment have hard-wired facility or share the same signals with SSLC.

Response:

1. The diverse I&C features are provided for protection against the common mode failures associated with the digital instrumentation and controls of the protection systems. The common mode failure primarily concerns the functions that are software-driven. The failure of a class 1E sensor is analyzed as a component failure. PSAR Figure 7.8-4 shows the output of sensors PT025, LT003, and FT009 are hardwired to their associated divisional RMUs, and simultaneously they are hardwired to PI625, LI603, and FI609 for indication. From the RMUs these signals are transmitted to the divisional SSLC DTM. The failure of any of these sensors will be detected at the SSLC DTM and the status of a failed sensor is available to the operator. The above hardwired indications provide a substitute for the digital software driven displays of the signals for the defense against common mode failures in the protection systems. Please refer to PSAR section 18.4, "Control Room Standard Design Features", and 18.7, "Human-System Interface (HSI) Design Implementation".

   No PSAR text change is required as a result of this response

2. The pressure display PI625, water level display LI603 or flow rate display FI609 will be accompanied by sound alarm when abnormal conditions occurs to pressure, water level or flow rate.

   No PSAR text change is required as a result of this response

3. The manual initiation signals for HPCF, RWCU, and RCIC start from their associated manual switches (by operator action) and are hardwired to their associated trip actuators which is a path that is completely different and diverse from the automatic initiation of these functions. The automatic initiation path start from the output of the sensors to the RMUs, SSLC DTMs, SSLC SLUs, and then to the trip actuators (see figure 7.8.4).

   No PSAR text change is required as a result of this response.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-033

PSAR Sections: 7.8.1.2

Question Date: April 4, 1998

PSAR Question:

1. Section 7.8.1.2 "Diverse Manual Controls and Displays" listed the diverse manual controls and displays of important functions. But when compared with K6/K7, Manual MSIV Closure, MSIV status display (Hardwired) and RPV pressure display (Hardwired) were not described. Please explain whether Lungmen has these designs? If not, what are the reasons?

2. Are all the Hardwired Manual Controls in the main control room grouped into one panel to facilitate operation?

Response:

1. PSAR section 7.8.1 (1) identifies the manual MSIV isolation as one of the diverse features for the defense against common mode failures in the protection systems. Also refer to PSAR section 18.4.2.5, "Fixed Position, Dedicated Function Switches", and note that Table 18.4-1, "Fixed Position Controls ", includes the Main Steamline Manual Isolation Switches.

   Table 18.4-3 , "Fixed Position Display Parameters", includes the MSIV position status (inboard and outboard valves) and RPV Pressure. Table 18.4 -5, "Fixed Position Alarms ", includes MSIV closure and RPV Pressure High. These parameters are also displayed at the Class 1E divisional visual display units (VDUs).

   No PSAR text change is required as a result of this response

2. All the hardwired manual controls are grouped in the Main Control Console (MCC). PSAR section 18.4.2.5 (second paragraph) states that : " the fixed position function switches on the MCC will be identified on the MCC panel Arrangement Drawing to be supplied with

the FSAR".

No PSAR text change is required as a result of this response

ROCAEC Review Comments:

TPC's response did not explain the question related to "RPV Pressure Display" in the original Question. Please supplement with further explanations.

Further Clarifications:

The second paragraph of the response to question 1 has been revised to address the RPV Pressure display.
No PSAR text change is required as a result of this explanation.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-034

PSAR Sections: 7.5.2

Question Date: April 3, 1998

PSAR Question:

PAM only has sections on the description of the system but no sections on the Specific Regulatory Requirement Conformance which makes it different from the other systems in the PSAR. Please supply such information.

Response:

Please see response to Question 07-003.

Section 7.5.1.1 provides an assessment of Regulatory Guide 1.97. Since the PAM is not a system in the same sense as the other systems described in Chapter 7, a subsection for "General Functional Requirements" would have no meaning, and the "Specific Regulatory Requirements Conformance" are covered in section 7.5.2.1. This is consistent with the format in the ABWR SSAR, which has been approved by the NRC.

In order to clarify this concept, the word "System" will be deleted from the PAM titles in section 7.5, consistent with its designation as "Post Accident Monitoring" in Table 7.2-1.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-035

PSAR Sections: 7.5.1.1 (PAM) (3) Table 7.5-1

Question Date: February 25, 1998

PSAR Question:

1. According to SRP, PAM should conform to RG1.151 - 1983 edition and RG1.153 - 1996 edition but commitments in Table 7.5-1 in PSAR 7.5.1.1 (3) can not conform to RG1.151 - 1983, RG1.153 - 1996 and the requirements of the quoted standards ANSI/ISA-S67.,01-1994, ISA-S67.,02- 01 - 1996, ANSI/ISA - S67.,10-1994, IEEE-Std384 - 1981, IEEE-Std603 - 1991, etc. Other sections on pressure regulation and sampling of the safety-related instruments have similar problems. Please correct them.
2. Similarly, Table 7.1-2 should be corrected too.
3. PAM system does not have "Specific Regulatory Requirements Conformance" section. Please provide such section.

Response:

1. Please see response to question 07-015 item g.
   Consistent with the ABWR Certified Design, the Lungmen NPS post accident monitoring (PAM) is not a designated "system" of itself. Rather PAM is accomplished by assimilation of a large set of discrete data, as defined by Regulatory Guide 1.97, which is gathered from the variables and parameters of many existing systems. The criteria identified in this question are addressed by the existing systems in accordance with the applicability required in Table 7-1 of the SRP. ANSI/ISA-S67.,01-1994 and ANSI/ISA - S67.,10-1994 do not appear in Table 7-1 or section 7.5 of the SRP and will not be addressed in the PSAR.

   No changes will be made to the PSAR as a result of the response to this question item.
2. See above item.
3. See response to questions 07-003 and 07-034.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-036

PSAR Sections: 7.5.2.3.2

Question Date: February 25, 1998

PSAR Question:

Lungmen PSAR chapter 1 Table 1.8 listed Codes and Standards are not consistent with SRP chapter 7 referenced Codes and Standards. For instance, IEEE-std338 was 1987 edition in SRP but in Lungmen PSAR it was 1977 edition; ANSI/ISA-S67.04 was referenced in SRP but not listed in Lungmen PSAR Table 1.8. Please review all the referenced Codes and Standards for their applicability.

Response:

Table 1.8-21 will be updated as follows:

- IEEE 338 will be updated with a year of "1987" to agree with SRP Appendix 7.1-C.
- The category of "Instrument Society of America (ISA)" will be updated to include ANSI/ISA-S67.04 with a year of 1994.

All referenced Codes and Standards contained in Table 1.8 will be reviewed for their applicability.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-037

PSAR Sections: 7.7

Question Date: February 27, 1998

PSAR Question:

RCIS multiplexing network consists of two separate channels, which utilizes fiber-optic communication links to transfer signals. Please describe in more details about the handling procedures in the situation when malfunction occur on any single or both channels, as well as the coordination of the two channels signals in normal and abnormal conditions.

Response:

For the normal RCIS operational case of no RCIS bypasses being activated, rod movement command signals and rod block condition signals are sent over both channels of the RCIS multiplexing system; and these signals are distributed to each channel of the associated Rod Server Processing Channels (RSPCs) associated with each FMCRD. Both RSPC channels associated with a given FMCRD send separate sets of control signals to the associated Inverter Controller and Rod Brake Controller. If a rod block condition is activated in either channel of logic, the rod block condition will be enforced (i.e. normal FMCRD motor movement is prevented). So, during normal operation, the rod block condition is enforced if either RCIS multiplexing channel transmits the signal corresponding to a rod block condition being active. This logic approach is already described in PSAR Section 7.7.1.2.1(5), on the top of Page 7.7-22, where it is stated: "If either channel or both channels of the RCIS logic receive(s) a signal from any of the following type of conditions, a rod block is initiated: ...". Even if no rod block condition is active, the associated rod movement control signals provided to each Inverter Controller and Rod Brake Controller (based upon signals transmitted via the RCIS multiplexing channels) are voted based upon two-out-of-two voting within the Inverter Controller logic and Rod Brake Controller logic. Therefore, both RCIS channels must transmit the same rod movement commands in order for control rod movement to be accomplished. This logic is also already described in PSAR Section 7.7.1.2.1(1), in the first complete paragraph of Page 7.7-17. This description only clarifies how the logic already described in the PSAR is applied for signals transmitted over the separate RCIS multiplexing channels (and subsequently transmitted to the Inverter Controllers and to the Rod Brake Controllers of the RCIS).

So, summarizing from the above, when no RCIS bypass conditions have been activated, the handling of signals sent via both RCIS channels is based upon using two-out-of-two logic for control rod movement logic and one-out-of-two logic for rod block logic (i.e. either channel can initiate rod block).

For the case of one or both RCIS multiplexing channelnetworks being failed, the RSPCs include logic (part of self-test logic) that senses failure of normal communications with the associated RCIS multiplexing channel and the RSPC logic (and associated Inverter Controller logic and Rod Brake Controller logic) automatically prevents any further rod movement in this condition, provided no RCIS bypass conditions are active. For this case, even if the one operable multiplexing channel sends valid movement command signals over one RCIS multiplexing channel, no rod movement would result. Similarly, if both channels are failed, further rod movement is prevented by the same type of logic. So, the RCIS logic is fail safe (i.e. no rod movement allowed) if one or both RCIS multiplexing channels fails and no RCIS bypass conditions exist.

For the abnormal case of only one RCIS multiplexing channel being failed completely, the operator can manually activate a RAPI channel bypass for the channel with the failed RCIS multiplexing channel to restore the operator capability to perform rod movement functions. The design of the RAPI bypass switch and the RCIS logic prevents the operator from bypassing both RAPI channels simultaneously. And as an additional protection feature in the RCIS logic, if the Inverter Controller logic senses that a a both channel bypass condition is being active, rod movement is prevented. Similarly the Rod Brake Controller logic assures the FMCRD brake will not be electrically energized if the both channel bypass condition is activated.

So, summarizing for the abnormal RCIS condition case, if one RCIS multiplexing channel fails and no RCIS bypass condition is activated, control rod movement by the RCIS is prevented. If one RCIS multiplexing channel is failed, it is possible to restore rod movement capabilities by activating a RAPI channel bypass for the failed channel. If both RCIS multiplexing channels are failed, rod movement capability is automatically lost and it is not possible to bypass both RCIS channels to restore rod movement capability in this case.

The existing PSAR description provides sufficient description of the basic logic used for signals provided from both channels of the RCIS multiplexing network. This response just provides a more detailed description of how this logic is implemented.

No PSAR text change is required by this response.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-038

PSAR Sections: 7.4.2.2

Question Date: March 2, 1998

PSAR Question:

7.5.2.1(2)(n) Suppression Pool Water Level is closely related to the core heat sink. Would it be more appropriate to classify it as Type B rather than Type C as listed? Please explain.

Response:

Per USNRC Regulatory Guide 1.97 Table 2 rev 3 the Suppression Pool Water Level is classified as a Type C. The purpose of this instrument is to detect a breach of the primary containment so it is a Type C.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-039

PSAR Sections: 7.6

Question Date: March 5, 1998

PSAR Question:

According to SRP Section 7.6 Areas of Review, this section should include :

1. interlock systems to prevent overpressurization of low-pressure systems when these systems are connected to high-pressure systems;
2. interlocks to prevent overpressure of the primary coolant system during low-temperature operation of the reactor vessel;
3. valve interlocks to ensure the availability of ECCS accumulators;
4. interlocks to isolate safety systems from non-safety systems (for example, seismic and non-seismic portions of auxiliary supporting systems);
5. interlocks to preclude inadvertent interties between redundant or diverse safety systems where such interties exist for the purposes of testing or maintenance.

But in Lungmen PSAR Section 7.6, only RHR High Pressure/Low Pressure Interlocks and Wetwell-to-Drywell Vacuum Breaker Interlocks are included and the later has title only and no text which when compared with SRP 7.6 Areas of Review, showed that PSAR Section 7.6 does not have the completeness and thoroughness. Please supply the information that is lacking.

Response:

The comments on Areas of Review for PSAR Section 7.6 that are listed above are addressed in the following manner:

1. An interlock used to prevent overpressurization of a low-pressure system (RHR) when it is connected to a high-pressure system (RCS)is discussed in Subsection 5.4.7.1.1.7 of the PSAR. Subsections 1C.2.45 105and 19.3.1.4.1 also discuss this topic.
2. Interlocks to prevent overpressurization of the primary coolant system during low-temperature operation of the reactor vessel are applicable to PWRs but are not used in the Lungmen NPS design and are therefore not discussed.
3. Valve interlocks to ensure the availability of ECCS accumulators are not used in the Lungmen NPS design and are therefore not discussed.

4. Information pertaining to interlocks that are used to isolate safety systems from non-safety systems are discussed in the following Subsections:
   - 9.2.13.2 (discussion of ECW interlock)
   - 9.2.11.1.1(3), 9.2.11.2, 9.2.11.3.2, and 9.2.11.5(discussion of RBCW interlock)
5. The topic of interlocks to preclude inadvertent interties between redundant or diverse safety systems where such interties exist for the purposes of testing or maintenance is not discussed since Lungmen NPS does not utilize these interlocks.

Section 7.6 of the PSAR will be updated to include appropriate cross-references to other sections, as indicated in Responses 1 and 4. Also, the section will be clarified to identify interlocks that are not applicable or not used, as indicated in Responses 2, 3 and 5.

ROCAEC Review Comments:

1. Even though the response indicated that the interlock of RPV low temperature over pressure is only applicable to PWRs so there is no discussion in PSAR but similar incident has happened in Kuosheng so please provide clarification as requested. If BWRs really do not need this interlock, should other means such as, administrative control or operator training, etc. be employed to properly deal with this concern?
2. The sentence at the beginning of the last paragraph "No changes to PSAR will be made as a result of the response to this Question" contradicts with the statement that followed and should be deleted.

Further Clarifications:

1. In addition to the interlock used to prevent overpressurization of RHR when it is interconnected to a high pressure system, prevention of potential overpressurization of the Reactor Pressure Vessel (RPV) during low-temperature conditions will be administratively controlled via TPC's Administrative Procedures and Operator Training. TPC's Lungmen NPS Administrative Procedures and Operating Training Course materials for operators will address the experience and lessons learned from the Kuosheng 1989 event wherein the RPV was pressurized at a low vessel metal temperature.
2. The sentence, as indicated in the comment, will be deleted from the response.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-040

PSAR Sections: 7.4.2.2

Question Date: March 2, 1998

PSAR Question:

The discussion section in Table 7.5-2 Post Accident Monitoring (PAM) System Variable List for items such as Neutron Flux, etc. were blank. Please explain.

Response:

The variables listed in Table 7.5-2 that have a blank discussion section are not described in section 7.5.2.1 are self explanatory and do not require further explanation.

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

ROCAEC Review Comments:

Do not accept the response. No matter the variable is clear enough or not, the major chapters and sections should be listed if it is discussed in PSAR for completeness.

Further Clarifications:

For the items discussed in other sections of the PSAR, a cross reference will be added to PSAR Table 7.5-2 in the upcoming PSAR amendment.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-041

PSAR Sections: 7.9

Question Date: March 2, 1998

PSAR Question:

Fiber Optics are widely used in Lungmen I&C system. Please provide the specification of the fiber optics used in NEMS and EMS. Also, in page 7-22 of NUREG 1503, GE committed to NRC that the EMS design will comply with ANSI ASC X3 T9.5(1998) but in Lungmen PSAR Table 1.8 no such code can be found. Please explain.

Response:

The fiber optics for EMS and NEMS will be specified for appropriate sheathing, pull strength, radiation resistance, fire retardance, and allowable line losses.

ANSI ASC X3T9.5 is not a standard; it is a subcommittee which had the task of examining and developing standards for FDDI physical, data link, and network layers. The actual standards for fiber optics specifications developed by ANSI ASC X3T9.5 are:

- ANSI X3.148 (1988): Fiber Distributed Data Interface (FDDI) - Token Ring Physical Layer Protocol (PHY)
- ANSI X3.166 (1990): Fiber Distributed Data Interface (FDDI) - Token Ring Physical Layer Medium Dependent (PMD)

These standards are listed in Lungmen PSAR Table 1.8-21

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: 07-042

PSAR Sections: 7.4.1.2

Question Date: March 4, 1998

PSAR Question:

Section 7.4.1.2.2 discussed the assumptions when main control room has to be evacuated and the 2nd item stated that when evacuating he plant is not experiencing any transient situations". Is this assumption reasonable? Please explain. For instance, a fire broke out in the control room and evacuation is required, then it is conceivable that part of the control equipment was damaged by the fire which caused plant transient. So, please evaluate the situation when evacuation is accompanied by transient and the effects on the design of the remote shutdown panel.

Response:

Main Control Room (MCR) evacuation due to MCR being uninhabitable is considered to be a rare (if ever) occurrence. Required MCR evacuation occurring during a plant transient occurrence is considered to be incredible. This is the type of plant transient situations assumed not to exist when the MCR becomes inaccessible. GE agrees that plant transients could originate as a result of whatever causes the MCR evacuation, as pointed out in this PSAR question. When the MCR must be evacuated, operations personnel are required to scram the reactor, to achieve initial reactor reactivity shutdown. Reactor scram by itself will cause transients. Remote Shutdown Panels (RSPs) will be analyzed to insure that they have adequate control/monitoring capability to achieve the cold shutdown condition, regardless of what plant transients originate as a result of whatever causes the MCR evacuation.

No changes to the PSAR are required as a result of the response to this Question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-043

PSAR Sections: 7.4.1.2.4

Question Date: March 5, 1998

PSAR Question:

Remote shutdown system has two panels. Are these two panels each capable to independently shut down the reactor? Are they in two different fire protection zones? Is the transfer switch which transfer the control when main control room is to be evacuated in the remote control room or the main control room? Please also explain under what conditions evacuations have to be done and what are the procedures for transfer of system control.

Response:

Each Remote Shutdown Panel (RSP) in itself is capable of an independent orderly shutdown.

The RSPs are to be located in separate rooms and in separate fire areas.

Each transfer device (switch) is to be located on the RSP that contains the controls/indications for the function being transferred.

Remote Shutdown System (RSD) design does not speculate what Main Control Room (MCR) conditions exist to cause evacuation such as fire,sabotage etc, however, RSPs are to be used for reactor shutdown when (if) the MCR becomes uninhabitable.

The first step in transferring control of a function from the MCR to the RSPs is to scram the reactor. The operator may manually scram the reactor before leaving the MCR, or the Reactor Protection System (RPS) logic input power breakers may be opened from outside the MCR as a backup means to achieve initial reactor reactivity shutdown. Once plant conditions are identified by checking the process variable displays available at an RSP, the operator will transfer control of the function to the RSP, whenever, the operator determines the need to do so, by means of a transfer switch. The actuation of the transfer switch overrides all associated control signals originating from the MCR, manual and automatic, thus transferring the function control point to the RSP.

No changes to the PSAR are required as a result of the response to this Question.

ROCAEC Review Comments:

1. Is there indication on the meters of the remote shutdown panel during normal operation?
2. It seems more appropriate to the response on the "first step" to verify that the remote shutdown panel is available before actual control is transferred.

Further Clarifications:

1. RSD indicators are active during normal plant operation.
2. Surveillance testing requirements assure with high reliability that the RSD panels will be available when needed. Also, the LCO assures that the RSD panel availability will be high. If the MCR becomes uninhabitable, timely MCR evacuation and transfer to the RSPs may be essential. Verification of RSP availability as a prerequisite to MCR evacuation and transfer to RSP would delay this process unnecessarily considering the high availability of the RSD panels.

The initial response to PSAR Question 07-043 is modified as follows to provide this needed clarification:

Single RSD panel shutdown capability is not a part of the RSD system design basis. See related response to PSAR question 07-044 for detailed information.

The RSPs are to be located in separate rooms and in separate fire areas.

Each transfer device (switch) is to be located on the RSP that contains the controls/indications for the function being transferred. The indicators are available for monitoring on the RSP during normal operation.

The Remote Shutdown System (RSD) design does not speculate as to what Main Control Room (MCR) conditions exist to cause evacuation such as fire, sabotage etc; however, RSPs are to be used for reactor shutdown when (if) the MCR becomes uninhabitable.

The Surveillance requirements of PSAR Chapter 16 assure a high probability that the RSD panels will be available when needed. Also, the RSD LCO (PSAR Subsection 16.3.3.6.2) insures that inoperable RSD panel functions are fixed in a timely manner.

The first step in transferring control of a function from the MCR to the RSPs is to scram the reactor. The operator may manually scram the reactor before leaving the MCR, or the Reactor Protection System (RPS) logic input power breakers may be opened from outside the MCR as a backup means to achieve initial reactor reactivity shutdown. Once plant conditions are identified by checking the process variable displays available at an RSP, the operator will transfer control of the function to the RSP, whenever the operator determines the need to do so, by means of a transfer switch. The actuation of the transfer switch overrides all associated control signals originating from the MCR, both manual and automatic, thus transferring the function control point to the RSP.

No changes to the PSAR are required as a result of the response to this Question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-044

PSAR Sections: 7.4.1.2.4

Question Date: March 5, 1998

PSAR Question:

Remote shutdown system has Div I and Div II panels but some systems do not have symmetrical inclusion on these 2 panels. For instance, the main steam system has 4 SRVs included in the remote shutdown system of which 3 are included in Div I panel and only one is included in Div II panel; also, FCS (Flammability Control System) has its control equipment included in Div II panel and not in Div I panel; N_2 (Nitrogen Supply System) system has its pressure indicator included only in Div I panel and not in Div II panel. Besides the above, are there any other layouts which are not symmetrical in the remote shutdown panels? And will any unsymmetrical layout hamper the capability of each individual panel to shutdown the plant? Please review in general terms.

Response:

Main Steam Safety Relief Valve (SRV) Nitrogen Supply Header Pressure indication is provided on both Remote Shutdown Panels (RSPs), because both RSPs have control/monitoring capability for at least one SRV. PSAR Section 7.4.1.2.4(13) will be modified to show the indication of Main Steam SRV Nitrogen Supply Header Pressure on both RSPs. Other unsymmetrical layouts found on the RSPs include:

a. The RHR WETWELL SPRAY LINE OUTBOARD PRIMARY CONTAINMENT VESSEL ISOLATION valve, 1E11-MBV-0026B, is controlled/monitored from the Division II RSP. There is no Loop A Residual Heat Removal System (RHR) equivalent valve to control/monitor from the Division I RSP.

b. The RHR DRYWELL SPRAY LINE CONTROL valve, 1E11-MCV-0028B, is controlled/monitored from the Division II RSP. There is no Loop A RHR equivalent valve to control/monitor from the Division I RSP.

c. The Loop B High Pressure Core Flooder System (HPCF) is controlled/monitored from the Division II RSP. There is no Loop A HPCF to control/monitor from the Division I RSP.

d. Although not needed for RSD, the Division I RSP contains two Reactor Core Isolation Cooling System (RCIC) indications to satisfy an RCIC requirement. There is no equivalent Loop B RCIC requirement to have indications on the Division II RSP. Therefore, there are no RCIC indicators mounted on the Division II RSP.

e. The RSPs contain indications to show the status of heating, ventilation, and air conditioning (HVAC) equipment serving areas containing safety-related equipment needed for remote shutdown. The only difference between the HVAC equipment that can be monitored from the two RSPs is the Division I RSP contains indications for the RCIC pump room air-handling unit (AHU) and the Division II RSP contains status indications for the HPCF pump room B AHU.

The RSPs will each be analyzed on their own merit to insure that they are independently capable of bringing the reactor to the cold shutdown condition. Even though the compliment of controls/indications on the two RSPs is not an exact match between the two RSPs, both RSPs will be equipped with adequate controls/indications to achieve and maintain the cold shutdown condition.

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

ROCAEC Review Comments:

(1) Why GE has unsymmetrical layout of the systems on the remote shutdown panels Div I and Div II ?

(2) How can one prove that each panel can bring the reactor to cold shutdown condition? Can someone provide a scenario where one division fails and the other can still shutdown the reactor ?

Further Clarifications:

The initial response to PSAR Question 07-044 is modified as follows to provide needed clarification:

Main Steam Safety Relief Valve (SRV) Nitrogen Supply Header Pressure indication is provided on both Remote Shutdown Panels (RSPs), because both RSPs have control/monitoring capability for at least one SRV. PSAR Section 7.4.1.2.4(13) will be modified to show the indication of Main Steam SRV Nitrogen Supply Header Pressure on both RSPs. Other unsymmetrical layouts found on the RSPs include:

a. The RHR WETWELL SPRAY LINE OUTBOARD PRIMARY CONTAINMENT VESSEL ISOLATION valve, 1E11-MBV-0026B, is controlled/monitored from the Division II RSP. There is no Loop A Residual Heat Removal System (RHR) equivalent valve to control/monitor from the Division I RSP.
b. The RHR DRYWELL SPRAY LINE CONTROL valve, 1E11-MCV-0028B, is controlled/monitored from the Division II RSP. There is no Loop A RHR equivalent valve to control/monitor from the Division I RSP.
c. The Loop B High Pressure Core Flooder System (HPCF) is controlled/monitored from the Division II RSP. There is no Loop A HPCF to control/monitor from the Division I RSP.
d. Although not needed for RSD, the Division I RSP contains two Reactor Core Isolation Cooling System (RCIC) indications to satisfy an RCIC requirement. There is no equivalent Loop B RCIC requirement to have indications on the Division II RSP. Therefore, there are no RCIC indicators mounted on the Division II RSP.
e. The RSPs contain indications to show the status of heating, ventilation, and air conditioning (HVAC) equipment serving areas containing safety-related equipment needed for remote shutdown. The only difference between the HVAC equipment that can be monitored from the two RSPs is the Division I RSP contains indications for the RCIC pump room air-handling unit (AHU) and the Division II RSP contains status indications for the HPCF pump room B AHU.

With respect to the shutdown capability of RSD, part of the design basis for the RSD system is that both RSD panels be used to control the full complement of RSD controllable functions and achieve the cold shutdown condition (refer to PSAR Section 7.4.1.2). This is the reason that there are instances of unsymmetrical layout of controlled equipment between the two RSD panels.

Although not part of the design basis, GE considers each RSD panel to have the capability to reach the cold shutdown condition (eventually) without use of the other RSD panel. Individually each RSD panel is capable of maintaining safe reactor core conditions due to the pressure reduction, water makeup, etc. capabilities that exist on each RSD panel. However, achieving cold shutdown conditions with only one RSD panel has never been fully analyzed by GE to insure that this capability exists, because this is not the

design basis. GE has not analyzed to determine scenarios where one division fails and the other can still shutdown the reactor because single RSD panel shutdown capability is not a part of the RSD system design basis.
As mentioned above, PSAR Section 7.4.1.2(13) will be modified as a result of the response to this Question.

ROCAEC Review Further Comments:

A fire in RSD panel room will cause one division to fail totally. In this case, GE should consider the capability to bring the plant to cold shutdown from the remaining RSD panel within 72 hours.

Further Clarifications:

GE notes that Standard Review Plan (SRP) Section 7.4 requires that the Remote Shutdown System (not just a single RSD panel) should be capable of achieving and maintaining cold shutdown conditions. The design basis for RSD does not require the adoption of the assumption that one panel of RSD is lost after the operators have been forced to evacuate the Main Control Room (MCR). The incorporation of this assumption would result in the system having to be designed for multiple independent and simultaneous failures (MCR evacuation plus RSD division failure).
The evacuation of the MCR represents a single failure. In such a case, both divisions of RSD are assumed to be available in order to bring the plant to cold shutdown conditions within 72 hours.
The total failure of one RSD division also represents a single failure. In such a case, the MCR can be assumed to be available to bring the plant to cold shutdown conditions within 72 hours, if necessary. If one division of RSD fails, the plant would enter a limiting condition of operation (LCO). If the failed division could not be repaired within 90 days, technical specifications would require that the reactor be shutdown.
For these reasons, the RSD design meets the regulatory requirements for single failures. Multiple failures are not a part of the design basis for RSD. Therefore, there is no requirement that an individual RSD panel be capable of reaching cold shutdown conditions within 72 hours.
A review of the various documents that address this subject revealed the need to clarify this requirement in the RSD System Design Description (SDD).
The first paragraph of the SDD, Section 3.3.5, will be rewritten as follows:
"RSD provides the capability to shut down the reactor from locations outside of the MCR, should the MCR become uninhabitable. This capability is

provided via two separate divisional control panels. Multiple means of achieving this system's objectives are designed into each of the two divisions. Transfer of control of the RSD panels isolates control signals from the MCR. This prohibits faulty signals originating from the MCR from adversely affecting RSD control.".

There will be no changes to the PSAR made as a result of the above response to the comments.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **07-045**

PSAR Sections: 7.5

Question Date: **March 10, 1998**

PSAR Question:

According to SRP Section 7.5 Areas of Review, this section should include :

- Post Accident Monitoring (PAM) systems;
- Bypassed or Inoperable Status Indication (BISI) for safety systems;
- Plant Annunciator (Alarm) Systems;
- Safety Parameter Display System (SPDS);
- Information System associated with the Emergency Response Facilities (ERF) and Nuclear Data Link (NDL).

But Lungmen PSAR Section 7.5 only includes Post Accident Monitoring System (PAM), Process Radiation Monitoring System (PRM), Containment Monitoring System (CMS), etc. and no descriptions on BISI and Alarm systems. Also, even though descriptions on SPDS and NDL were said to be combined with Section 7.7.15, but in 7.7.1.5 relevent descriptions were lacking. Please explain or provide information that are lacking.

Response:

Information that deals with Bypassed or Inoperable Status Indication (BISI) for safety systems is discussed for each individual safety system via the subsection that discusses its conformance with Regulatory Guide 1.47. An addition to Section 7.5.1, 2nd paragraph, 1st sentence will be made as follows: "Bypassed and Inoperable Status Indication for appropriate safety systems is discussed under each individual system via the topic of Regulatory Guide 1.47."

Annunciation of alarms is discussed within Subsection 7.7.1.5 "Process Computer System (PCS)" and its conformance to the applicable requirements is enveloped within the discussion for PCS that is contained in Subsection 7.7.2.5. A new sentence will be added, for clarification purposes, in

Subsection 7.5.1, 1st paragraph immeditately prior to the next to last sentence stating: "Information pertaining to Plant Annunciation (Alarms), which are part of the Plant Computer System (PCS), can be found in Subsections 7.7.1.5 and 7.7.2.5."

Information pertaining to the Safety Parameter Display System (SPDS) can be found in Sections 18.4.2.7(2) and 18.4.2.10. Section 7.5.1, 1st paragraph (last sentence), will be modified to state: "The SPDS functions, are discussed in Subsections 18.4.2.7(2) and 18.4.2.10. Please note that, in the ABWR control room design, the information required for a Safety Parameter Display System is included in the basic set of information presented to the operators on the Wide Display Panel portion of the Main Control Room Panels."

The Emergency Response Facilities (ERF) and the Nuclear Data Link (NDL) are implemented in the Lungmen design via an onsite Technical Support Center (TSC), nearsite Emergency Operations Facility (EOF), and Emergency Executive Committee (EEC) facility.

ROCAEC Review Comments:

ERF and NDL are not include in the responses. Please supplement.

Further Clarification:

1. A new subection **7.5.3 "Emergency Response Facilities (ERF) and Nuclear Data Link (NDL)"** with the following text will be added to Chapter 7:

"**7.5.3 Emergency Response Facilities (ERF) and Nuclear Data Link (NDL)**

In the onsite TSC, nearsite EOF and the EEC (which is located at TPC Headquaters), workstations (a total of two at each location) with 21" non-touch screen CRT, a laser printer and a color copier are provided to perform monitoring of the plant for technical support. These workstations are connected to the PCS LAN. A modem is used for the connection between EEC and the PCS LAN. For further detailed description, please refer to

Appendix C "Emergency Plan", Section C.4.1 and C.4.2."

2. Change subsection 7.5.3 "Reference" to subsection 7.5.4"Reference" 。

PSAR will be modified as described above 。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track　Number)：　07-046

問題章節(PSAR Section)：　Chapter 7

初提日期(Question Date)：　1998.5.27

問題內容(PSAR Question)：

1. 儀控系統宜考慮Distributed方式，然後用網路連起來，並儘量用光纖以避免EMI；另外廠內各單位及與廠外相關單位間相資訊之傳輸，宜採類似Internet, Intranet, Extranet的技術，力求通訊協定標準化即可。
2. 請參採AT&T電話系統之Cold Stand-by Computer, Hot Stand-by Computer的技術，以提昇儀控系統可用性。
3. SENSOR出來的小訊號宜就地放大處理，經數位化後變成標準介面才送出去。
4. 定期自動利用空檔時機，自我偵測診斷系統是否正常；儀控系統軟、硬體教應儘量模組化，並考慮將來的可更換性。
5. 各儀控系統及設備必須合乎EMI/EMC及FCC等的管制規格。
6. 儀控系統之維護宜參考民航機維修制度，譬如定期更換等。

問題答覆(Responses)：

1. 儀控系統採用以光纖作為通訊連結方式之網路架構，在可行範圍內儘可能應用網路技術(例如，使用標準通訊協訂)。然而，安全有關控制系統因有控制環境之安全顧慮，故必須限制採用完全開放之架構。安全與非安全有關控制系統之間，以通訊閘作為隔離之介面裝置。
2. 廠用電腦系統(PCS)屬非安全系統，該系統不引發或控制任何安全裝備。所有Class 1E裝置之控制均符合所需之控制可靠度。
3. 問題中所謂微弱信號如指RTD、Thermocouple等所產生之微弱電壓/電流信號，此類信號之放大與處理將儘可能靠近信號源，以確保其

I/O裝置接收到之信號不受干擾。有些應用可能須要在感信號點即作信號處理。如果信號點就在遠端多工裝置(RMU)或輸出入卡片附近，這類信號在轉換為數位信號之前不須作信號之放大。

5. 問題中所述之功能均已納入設計，請參考PSAR 7.1.2.1.6節。

   易受EMI/EMC影響之設備係根據PSAR 7.1.2.1.7所列之標準設計，但聯邦通訊委員會(FCC)之標準則不適用於核能電廠儀控設備。

6. 奇異將根據設計壽命長短而分類(元件及子系統)建立設計壽命分類系統(Design Life Classification System)及清單。此系統將被納入於預防保養及檢視/更換計畫內。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-047

PSAR Sections: 7.8

Question Date: May 11, 1998

PSAR Question:

1. Does the initiating event of common mode failures include only the SSLC common mode failures or does it cover all NUMAC system which includes NMS?
2. If the initiating event covers all NUMAC system, then what is the development of the resulting transient scenarios? According to the current Lungmen design, is it possible to manually activate the whole SLC system through the SLC Keylock Switch?
3. Also, what are the conclusions of the analysis report of the initiating events and its transient scenarios of Lungmen Essential Multiplexing System failure?

Response:

1. The NUMAC instruments, when used for SSLC and NMS, are configured differently (with unique signal conditioning and processing components for each application) and have different application software. However, the CPU and operating system are the same; thus a common mode failure could affect all systems using NUMAC (SSLC, NMS, Essential Multiplexing System (EMS), Process Radiation, ATIP). It is assumed that a common mode software failure due to initial software design or an unanticipated unintended function would be the initiator, since a common mode hardware failure is not considered credible.

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

2. Even if all NUMAC equipment is affected simultaneously, the effects on transient scenarios and safety functions are the same. NMS is only one of several diverse trip inputs to SSLC. An SSLC or EMS common mode failure due to NUMAC failure has a global effect, disabling almost all automatic trip functions. However, this condition has been

analyzed and the resulting diversity and defense-in-depth features have been incorporated into the safety system design as described in PSAR Section 7.8. If the common mode effects are not simultaneous, then the continuous self-diagnostics in each NUMAC instrument would be expected to mitigate common mode failure effects by detecting the failure and permitting the operator to take appropriate action (bypass, trip, etc.).

In the current Lungmen design, it is possible to manually activate the whole SLC system with the keylock switches (Div. I and Div. II). Please note that this action is performed in separate and diverse circuitry from the RPS NUMAC portion of SSLC and is not affected by the same common mode failure effects.

The SLC system is not interlocked with the SRNM ATWS permissive signal. It is possible to initiate the SLC System through the Keylock switch if the "SRNM" Permissive signal is lost due to all NUMAC equipment failing simultaneously. The SLC System Keylock switches are separate from the Manual ATWS mitigation Push Button Switches. SLC Keylock switches are located on the left side on Main Control Console Apron and the ATWS mitigation Push Button Switches are located on the middle of the Main Control Console Apron.

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

3. Initiating events and transient scenarios for Lungmen EMS are discussed in Appendix AI.6 of the PSAR. GE has committed in a previous response (Track No. 07-012) to update the diversity and defense-in-depth study for the FSAR to include Lungmen-specific features and the final configuration for EMS and SSLC. The results of this study will be factored into the PRA.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-048

PSAR Sections: Ch. 7

Question Date: May 18, 1998

PSAR Question:

1. What are the design, operation and functional requirements of feedwater control system LFCV? Please explain how these functions can be accomplished. Is the design of LFCV consistent with the requirements of PSAR Appendix E on experience feedback?

2. Please clarify whether the design meets the requirements of seismic qualification and piping configuration. Also, has its natural frequency been evaluated to make sure its operation and startup will not be adversely affected?

3. Please compare the pros and cons of LFCV design between Lungmen and Kuosheng NPS.

4. In PSAR Figure 10.1-1 when switch between bypass line and startup of motor driven FW pump, is there any transient state caused by this switch? If there is, please explain its impact.

PSAR Response:

1. The design requirements (pressure, temperature, flowrate) of the LFCV are consistent with the portion of the feedwater system that the low flow control valve is installed in.

   Operational requirements of the LFCV include

   - During plant startup, in conjunction with the MDRFP to control reactor level.
   - During plant transients (i.e. trip of one TDRFP) in conjunction with the automatic start of the MDRFP and in parallel with the operating TDRFP to control reactor level based on signals conditioned proportionally to the MDRFP flow rate.

   Functionally the LFCV must supply ~2 to 20% NBR feedflow for both cold water and normally operating feedpump temperature water. The 2% low limit represents the minimum "bias" that will be generated by the APR's startup program opening of the bypass valves (to prevent too low a flow/stratification/wire drawing) and the 20% represents the motor feedpump capacity.

Differential pressures will be the difference between pump head/flow curve and reactor pressure for the applied load.

The experience feedback will be satisfied by reviewing vendor information on similar applications of the valve (i.e. similar dPs) during the procurement/ purchase process.

2. Seismic requirements are specified in the valve procurement specification. The piping system and valve are designed to ROC Building Code requirements. Natural frequencies as they affect startup and operation are not specifically evaluated in the design phase but adequate operation will be verified during startup testing.

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

3. There are neither pros nor cons of the LFCV between the two plants. Comparisons can be made of the design features, operating requirements and/or functions if required. However, the concept of the LFCV and proposed application is not unique to either plant. There are no advantages or disadvantages to this design that would be considered special or would require special provisions to implement.

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

4. Note that this figure no longer applies to the Lungmen FW system design and will be updated (or deleted as applicable) for the Lungmen FSAR.

   A very small flow transient is expected. It is anticipated that this transient will cause only a negligible level spike and flux spike.

Generically this concern is mitigated by the following design features:

- Between 2 - 5% of rated FW flow will be passing through the valve at the time the switch occurs.
- The reactor will be steaming at 2 - 5% of rated steam flow, and the corresponding voids will make the reactor much less susceptible to cold water spikes (also RWCU flow will kept as high as possible, within equipment design constraints, to minimize the perturbation.)
- A very slow 1- 2 min MDRFP discharge valve has been specified, with the added requirement that the valve be as linear as possible (e.g. globe instead of gate).
- The FW control system will be optimized to respond to increased dP across the LFCV when the motor driven FW pump starts.

原能會審查意見(ROCAEC Review Comment):

1. 可接受。
2. 審查中。
3. 仍請台電提供龍門與國聖 LFCV 設計優劣比較。
4. 可接受。

台電澄清說明(Further Clarification)：

LFCV設計優劣會隨飼水控制系統之架構而改變，現將龍門(核四)與國聖(核二)所應用之功能說明如後：

1. 龍門飼水系統中LFCV是位於馬達驅動飼水泵(MDRFP)之出口，於低飼水流量時來控制Rx水位，當Rx加熱與增壓時會有部分蒸汽持續被排放至冷凝器(約4%)，此時Rx之水位需要被控制在所需之正常水位設定點，直到汽機驅動飼水泵(TDRFP)起動，飼水流量增至15%時，漸漸將流量由TDRFP取代而將LFCV慢慢關閉後MDRFP停止，此時飼水流量已增至25%預備起動第二台TDRFP而完成低飼水流量之控制。(請參閱附圖一，龍門電廠飼水系統P&ID)
2. 國聖飼水系統中沒有MDRFP，其LFCV有二只，其一是位於三台TDRFP出口集管下游(LV-241)，另一是串聯於旁通飼水泵管路上(LV-242)，當Rx於加熱

與增壓時，由冷凝水泵旁通TDRFP經LV-242來建立水位控制。當Rx壓力上升至冷凝水泵已無法提供足夠之水頭壓力時，旁通飼水泵管路關閉，第一台TDRFP起動，此時飼水流量經由LV-241來建立水位控制，直到流量增加需由第二台TDRFP加入後完成低流量之控制。（請參閱附圖二，國聖電廠飼水系統P&ID）

3. 基本上LFCV之功能在二個電廠中是一致的，均屬低流量時用以控制Rx於正常水位，國聖廠因沒有MDRFP所以在Rx增壓的過程中需較早由TDRFP起動建立較高的水頭壓力，初期並需由另一控制閥(LV-241)來控制飼水流量，而龍門廠從Rx開始加熱至第二台TDRFP起動前均由LFCV來完成水位控制，其優點是除涵蓋國聖廠原有之特性外，控制上由啟動到低功率運轉均由單一元件來達成，避免增加更換元件動作增加運轉員之負擔而影響系統之穩定性。

4. 根據國聖廠運轉程序書說明，20%功率前是由LV-241控制，20~30%由主控制器自動控制TDRFP轉速，以維持反應爐水位。而龍門之低流量控制以目前之設計約為15%流量（尚未定案）。因為MDRFP飼水容量為20%，而飼水流量之量測會有些許誤差，且在運轉時模式之變更一定會有重疊(overlap)之情況，故將低流量控制模式定在約為15%流量應屬合理（LFCV有25%之容量）。

   模式變更之切入點對控制器而言並非主要考量之因素，況且龍門採用三重容錯數位式控制之設計（FTDC），優於國聖廠所用之傳統飼水控制系統，而使水位控制更加穩定。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

編號(Track Number)：　07-049

問題章節(PSAR Section)：　Chapter 7

初提日期(Question Date)：　1998.5.22

問題內容(PSAR Question)：

1. 核四採用Digital I&C方式作為電廠操作的控制方法，請提供日本與英國電廠採用 Digital I&C 的經驗與核四廠不同之相關資料。
2. 核四廠採用按鈕控制會較以往電廠產生更多之電磁波，核四廠對於電磁波之減少干擾是否有特殊的因應措施，請澄清。
3. 核四廠是否有考量數位儀控系統與其他控制系統的界面整合，請澄清。
4. Digital系統與Analog系統其失敗的可能性並不相同，軟硬體亦不同。是否會因此造成測試計畫的變更，請澄清。
5. 數位化儀控系統之電源系統是否與其他系統隔離，以避免受到暫態實波(Spike)干擾，請澄清。
6. 未來核四廠人員訓練計畫建議應特別著重數位化儀控系統的操作及軟硬體維護能力。
7. 在電廠的運轉生命週期內其控制系統的軟體維護計畫如何執行請提供相關資料。

問題答覆(Responses)：

1. GE已準備一份簡報資料，並已於七月十日向原子能委員會報告有關Sizewell B/Chooz B-1經驗及與龍門儀控系統之比較。

2. 易受EMI/EMC影響之設備將根據PSAR 7.1.2.1.7節中所列之標準設計，其中包括雜訊遮敝電纜之使用及大量使用光纖電纜。

3. 數位系統與其它控制系統之界面整合詳如PSAR Table 7.1-1及Figure 7.1-1說明。

4. 根據RG1.168規定，應用在儀控系統之軟體均需有驗證與確認計畫(SVVP)。此計畫應定義以軟體為主之重要系統所須執行之相關驗證與測試需求。

   基本上，數位系統之測試方案與類比系統並無不同。此方案應包括在安裝、試運轉及功率運轉等測試之前之驗證測試;在設備裝船交運之前,尚須執行一項所謂在廠整合測試。然而，有部份在廠整合測試項目在設備於現場安裝完成後，可以併同試運轉測試再次執行。重複執行之項目應依據 SVVP 中驗證原始設計之適用項目決定。試運轉測試之項目至少應包括感測器控道校正與檢視、分區功能測試、邏輯系統功能測試、輸出控道功能測試、整體功能測試。功率運轉測

試用來驗證以軟體為主之儀控系統功能與設計相符合。那些在試運轉與功率運轉測試中重複執行之 V&V 項目，將被成為技術規範中(Technical Specification)電廠壽命期內之定期測試與維護測試項目。

5. 電力系統與安全系統邏輯/控制(SSLC)間之界面說明如PSAR 7.1.1.2.1.2節所述。每一個數位儀控系統均使用各別之電力供應，此電力由120AC或125DC轉換成低直流控制電源而得。這些電源具有需要之隔離，以抑低系統對電源端產生之暫態突波之敏感度。非安全控制系統電源由Non-1E電力系統供電，此電源與安全控制系統之電源完全獨立、分離。
6. 龍門人員訓練說明詳如PSAR 13.2節。這些由臺電規劃之訓練項目包含數位儀控系統軟硬體操作與維護訓練。
7. 軟體維護屬標準軟體發展生命週期(Standard Software Development Lifecycle)內之項目之一，在龍門軟體發展計畫(軟體管理計畫、軟體構型管理計畫、軟體驗證與確認計畫)內有完整記載。這些計畫記載關鍵維護項目需執行之作業以及應遵循之程序，例如失效報告、失效原因分析與追蹤、紀錄文件保存及後續作業以及修改軟體之構型管理程序等。

原能會審查意見(ROCAEC Review Comment) ：

請台電公司針對如何確保軟體維護品質，提供較詳細說明資料。

台電公司澄清說明(Further Clarification)：

1. 軟體維護屬標準軟體發展生命週期 (Standard Software Development Lifecycle) 內項目之一，在龍門軟體發展計畫 (軟體管理計畫、軟體構型管理計畫、軟體驗証與確認計畫及軟體維護計畫)內有完整記載。這些計畫記載關鍵維護項目需執行之作業以及應遵循之程序，例如失效報告、失效原因分析與追蹤、記錄文件保存及後續作業以及修改軟體之構型管理程序等。
2. 核電廠依規定平時需執行之維護測試是定期測試 (Surveillance test)及自動偵錯測試 (Automatic diagnosis test) 是屬系統測試 (軟、硬體一起)及組件測試 (硬體)。軟體不像硬體，它永遠不會被磨損折舊，因此與硬體維護相關的主要行為－更換破損部份－將不會出現。在運轉與系統測試中若發現軟體本身設計錯誤，將依軟體維護計畫及手冊執行更正。
3. 另依法規要求增加之功能需求以及軟體版次更新等將依軟體維護計畫執行修改需求之確認、可行性及安全分析、軟體設計、軟體製作、系統測試、驗收測試等程序，並將上述之每階段產生文件納入構型管理。
4. 目前台電有關儀控軟體維護人員之培訓從民國八十六年開始每年派二人每人各半年赴原廠家 (GE 公司)執行有關軟體維護方面之技術移轉以及參與儀控軟體

設計工作，此項計畫將持續至九十年。另將來電廠在運轉前將擬定一整體軟體維護計劃，此計劃涵蓋維護團隊之組成、人員之訓練、技術之引進、軟體構型管理之維護及原廠技術支援之介定等，以因應運轉時執行軟體故障之維修，此將可建立軟體維修技術之自主能力。另合約規定，在電廠運轉後三年內為保固期， GE 仍擔負正常運轉之責任。同時，由過去核一、二、三廠之經驗對於相關之軟體維護除台電外，亦視不同計畫與需求將請原設計廠家或國內外之專業公司之技術支援（國外如 GE 、西屋、 SAIC 、 MPR　A 國內如核研所、資策會、新鼎、公元等）。尤其特別在核四計劃，透過「工業合作協定」，要求原廠家 GE 提出一定額度的技術、設備轉移至國內相關企業單位。目前核研所已引進原廠家有關核四儀控軟體設備與技術，對將來台電之軟體維護更易落實於本土。

5. 軟體在設計發展過程 (從系統需求、軟體規範、軟體設計、編碼、測試、驗收測試)中所產生之軟體文件以及設計過程中所發生失效原因、失效分析與修正情形之所有文件皆透過構型管理 (configuration management)來保存，可使將來在執行軟體維護時可透過構型管理文件追蹤，不會因原設計廠家或台電公司相關人員調離而造成軟體無法維護狀況。另在軟體維護程序上亦應遵循相關軟體維護標準規定來執行，以確保軟體維護之完整與正確。

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-050

PSAR Sections: 7.1.1.2.1.3

Question Date: June 20, 1998

PSAR Question:

Many sensor signals are shared by the control logic of multiple safety systems, thus significantly reducing the number of safety-related sensors in comparison to conventional plants. Please explain in detail which sensors will be shared by which control logic of multiple safety systems. What is the applicability of this shared sensor design? Please also analyze the potential consequence from failures of the shared sensors. Is there any possibility of common mode failures?

Response:

1. **Shared sensors**
   - Main Steam pressure and level sensors for RPS and ESF
     - narrow range RPV water level
     - wide range RPV water level
     - RPV dome pressure
     - drywell pressure
   - Suppression Pool Temperature Monitor (SPTM) bulk average temperature
   - MSIV closure (also hardwired to Steam Bypass and Pressure Control system)
   - Condenser vacuum

2. **Applicability of shared sensor design**
   To reduce plant construction costs and simplify maintenance activities, the ABWR protection systems are designed with a "shared sensors" concept. Safety System Logic and Control (SSLC) is the central safety system logic processing mechanism and produces logic decisions for both RPS and ESF safety system functions. Redundancy and single-failure requirements are implemented by a 4-division modular design using 2-out-of-4 voting logic in each electrical division on inputs

derived from signals measuring diverse parameters (e.g., reactor water low level and high drywell pressure). Many additional signals are provided, in groups of four or more, to initiate RPS scram as shown in Item 1.

3. **Consequences of failures of shared sensors**

The shared sensor concept is feasible because of the 4-division, bypassable 2-out-of-4 configuration of the digital protection system. Random single failures of shared sensors have no effect on safety since the system reverts to 2-out-of-3 of the remaining sensors when a failed division is bypassed. Note that a half-scram condition does not occur. If another failure occurs in a remaining channel, that channel can be manually tripped. A trip condition in one of the two remaining channels will then trip the plant.

A diversity and defense-in-depth analysis as required by the revised SRP, Chapter 7, has already been performed for the US certified ABWR design on several possible different configurations of SSLC. This analysis will be updated for the Lungmen-specific design, but the conclusions are not expected to change since the multi-redundant, voted coincident logic concept is the same. Analyses have been performed at the system design level to assure that adequate defense-in-depth and/or diversity principles were incorporated. It is recognized that such requirements are in addition to positions on safety-related protection systems (such as the single failure criterion) taken previously in various Regulatory Guides.

4. **Possibility of common mode failure**

With its inherent advantages, it is also recognized that such design integration (i.e., shared sensors) theoretically escalates the effects of potential common-mode failures (CMF). CMF of sensor and transmitter hardware is considered to be a very low probability. However, SSLC system architecture is designed to provide maximum separation of system functions by using separate digital trip modules (DTMs) and trip logic units (TLUs) for RPS/MSIV logic processing and for LDI/ECCS logic processing within each of the four essential power divisions. Thus, setpoint comparisons within individual DTMs are associated with logically separate initiation tasks.

Sensor signals are sent to each DTM on separate and redundant data links such that distribution of DTM functions results in minimum interdependence between echelons of defense. For reactor level sensing, the RPS scram function utilizes narrow-range transmitters while the ECCS functions utilize the wide-range transmitters. The diverse high drywell signals are shared within the 2-out-of-4 voting logic. In addition, all automatic protective functions are backed up by manual controls.

Shared sensors are under both automatic and manual periodic surveillance to confirm that the redundant sensors are operable and reading within a reasonable tolerance of each other.

As a general rule, shared sensors for protection systems are not used for control systems (i.e., feedwater, recirculation flow control, etc.). However, the end-of-cycle (EOC) recirculation pump trip (RPT) signals originate from the same turbine stop valve closure or turbine control valve fast closure sensors which contribute to scram. These are Class 1E sensors, but the EOC-RPT signal is sent via fiber optic cable (for isolation) to the recirculation flow control system. No signals are sent from the control systems to the safety-related systems.

Another use for some of the protection shared signals involves the ATWS trip which activates the Fine Motion Control Rod Drive (FMCRD) run-in and alternate rod insertion (ARI) as diverse backup to hydraulic scram. However, this Class-1E-to-non-Class-1E isolated interface is a special case for mitigation of ATWS and is not a control system interface. These signals are also processed as 2-out-of-4 and are bypassed together with the SSLC functions.

As a final backup for CMF, one division of manual HPCF start is hardwired around SSLC along with hardwired RPV level and drywell pressure indicators for operator information. The Remote Shutdown System (RSD) also provides an independent means of actuating core cooling functions diverse from the plant main control room.

The Lungmen PRA takes into account the sharing of sensors among the safety systems. The interdependencies that could result in CMF are included in the model.

In summary, the ABWR design has incorporated defense-in-depth principles through maintaining separation of control and protection functions even though sensors are shared within protection systems. Diversity principles are incorporated at both the signal and system levels: (1) diverse parameters are monitored to automatically initiate protective actions which are also manually controllable; and, (2) multiple diverse systems are available to both shut down the reactor and to cool its core.

No changes will be made to the PSAR as a result of the responses to the Questions associated with Track Number 07-050.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 07-051

PSAR Sections: 7.4

Question Date: June 22, 1998

PSAR Question:

PSAR Section 7.4 "Safe Shutdown Systems" does not include the description of Alternate Rod Insertion (ARI) function. However, GE SSAR Section 7.4 has description related to ARI. Please explain why ARI description is not included in PSAR Section 7.4. Also, state whether ARI has any safety-related functions.

Response:

The ARI function is discussed in PSAR Section 7.8 (i.e., 7.8.1.2 and 7.8.2.2). This is because the PSAR format is organized based on the newer version of the SRP, which provides the new Section 7.8 for "Diverse Instrumentation and Control Systems". However, the GE SSAR was organized based on an earlier version of the SRP, which did not have Section 7.8. Therefore, the ARI was placed in Section 7.4 for the GE SSAR. It is appropriate that it be included with Section 7.8 in the PSAR because the function is specifically designed to provide a diverse means of inserting the control rods.

There are no safety-related functions associated with the ARI.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-08-001

PSAR Sections: 8.1.2.2; 8.3.1.3.1

Question Date: Dec. 5, 1997

PSAR Question:

Please clarify the following questions regarding Emergency Diesel Generators:

1. If Bus A4, B4 and C4 "normal preferred power" tripped and "alternate preferred power" successfully come on line, will Emergency Diesel Generators automatically start ?
2. Emergency Diesel Generators can be started or stopped in the field. Is it possible to synchronize it to the grid in the field ?
3. Why it takes 20 seconds for the Emergency Diesel Generators to go from emergency start to rated frequency and voltage ? What are the design bases ?

Response:

1. The emergency diesel generators are not anticipated to start if the buses successfully transfer to their fast transfer source. If the fast transfer source is not available or if the transfer fails the bus voltage will drop to zero and the EDG will start.
2. Manual EDG synchronization and initiation of automatic synchronization may be performed from the main control room or locally.
3. The 20-second timing is the MAXIMUM time allowed for the EDG to reach rated voltage and frequency, and to energize its bus so that the Class 1E load sequence can begin. In other words, the EDG is specified to energize its bus in LESS THAN OR EQUAL TO 20 SECONDS. This is based on the analysis performed for the emergency core cooling systems start times in order to ensure adequate core coverage.

   In actual operation, according to manufacturers, the EDG may be expected to energize its bus in about 10 seconds or less. However, experience has shown that it is best to allow the EDG to start at its own pace, within allowable limits. Forcing the diesel to start earlier than necessary (particularly for the many start tests it must undergo), reduces its reliability and design life.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-08-002

PSAR Section: 8.1.4

Question Date: December 5,1997

PSAR Question:

Based on one-line diagram (8.3-1), the sequence of Class 1E Bus A4, B4 and C4 energizing and switching are as follows:

Bus	1-(Normal Preferred power)	2-(Auto Transfer Alternate Preferred power)	3-(Manual Transfer Alternate Power)
A4	UATA	RAT2	RAT1
B4	RAT2	UATB	RAT1
C4	UATC	RAT2	RAT1

Therefore, should Section 8.1.4 on page 8.1-11be amended as the following:

1. As indicated in Section 8.1.4, line No. 10 from bottom, " .....(Main, UAT A, B, and C for Class 1E A4 and C4 and RAT2 for Class 1E B4)" should be changed to " .....(Main, UAT A and C for Class 1E A4 and C4 and RAT2 for Class 1E B4)" . Main, UAT B in the above sentence should be deleted.
2. As indicated in Section 8.1.4, line No. 7 from bottom, ".....RAT2 also supplies manual fast transfer....." should be changed to " RAT1 also supplies manual transfer....." .

PSAR Response:

The following changes will be made to PSAR Section 8.1.4:

1. Section 8.1.4, paragraph 4 - .....loss of normal preferred power (Main, **UATA, and C** for Class 1E bus A4 and C4 and RAT2 for Class 1E B4)."
2. Section 8.1.4, paragraph 5 - .....RAT2 also supplies manual fast transfer alternate preferred power to Class 1E bus **A4, and C4**. RAT2 also supplies normal preferred power to the Class 1E bus S4 (the swing emergency diesel generator bus). **RAT1 supplies manual transfer alternate preferred power to Class 1E bus A4, B4, and C4**."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-08-003

PSAR Section: 8.2.4.2

Question Date: December 5,1997

PSAR Question:

1. Load rejection - If a fault occurs on 345 kV transmission lines, can the main generator be operated to supply power to plant auxiliary system alone?.
2. Gas Turbine: Why is it not considered as power source for station blackout condition ?

PSAR Response:

1. The consequences of any fault on one of the four 345 kV transmission lines or any fault on one of the two 345 kV buses are stated in PSAR Table 8.2-1. In case of multiple faults on all 345 kV transmission lines, the main generator will lose its major load, and the plant will be operated under full load rejection scenario. The main generator is capable of supplying power to the plant auxiliaries only.

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

2. Lungmen Nuclear Power Station does not employ gas turbine as an Alternate AC power source for Station Blackout condition. The swing diesel is used for this function.

   The swing emergency diesel generator is designed with diverse features and has sufficient capacity and reliability for operation of all systems required for coping with station blackout. It will be designed to operate for the time required to bring and maintain the plant in safe shutdown (non-design basis accident). Thus, it will fill the role of an alternate AC power source, as defined in 10CFR50.63 (c) (2) and Reg. Guide 1.55.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-08-004

PSAR Sections: 8.3.1.1.1

Question Date: December 5, 1997

PSAR Question:

1. The fine motion control drive system is graded class 1E, whereas the motor is not. This is an inconsistency. Please clarify.
2. Since C3 bus supplies the fine motion control drive system, why isn't there a "fast transfer" design (ref. Fig. 8.3-1) for C3 bus?

Response:

1 The FMCRD system is not safety related, however the primary source of power is Class 1E (i.e., Bus A4). The circuits from the Class 1E medium voltage switchgear down to the FMCRD transfer switches are Class 1E associated and are treated as Class 1E circuits. The circuits and equipment downstream from the transfer switches are all non-Class 1E.

2. As indicated in Response 1, the primary source of power to the FMCRDs is the Class 1E bus (A4), which is backed by the emergency diesel generators. The C3 bus is the secondary source, and is only used for the FMCRD (via the transfer switch) if the primary source fails. Hence, the FMCRDs have sufficient power source redundancy and do not require fast transfer to the secondary source.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track No.: **I-08-005**

PSAR Sections: **8.0/8.1**

Question Date: **January 2,1998**

PSAR Question:

1. Please mark the rated capacity (KVA or MVA) for Main Transformer UATs & RATS.
2. Paragraph. 8.1.1, item (7) - Please explain more specifically the meaning of "stuck circuit breaker will not trip more than one additional unit or line".
3. Both types of (1) Solid State relay (SS Ry) and (2) Magnetic relay (EM Ry) are usually required to be implemented in relay protection system design to meet the redundancy and diversity criteria. Descriptions in the PSAR are not clear regarding this issue, please describe more specifically.
4. The following features are required in Relay protection system design:
   (1) Phase to Phase fault protection
   (2) Single phase ground fault protection
   (3) Bus Bar protection
   (4) CB fault protection
   (5) Re-closing relay
   Descriptions in the PSAR are not clear regarding this issue, please describe more specifically.
5. ABS shall be equipped with the following features:
   (1) Motor operate mechanisms with electrical interlock,
   (2) Manual operate mechanisms and
   (3) Grounding switch.
   Please clarify.
6. Is SWYD indoor type? Please clarify.

PSAR Response:

1. The main transformer sizing calculation is performed by S&W. The UAT & RAT transformer sizing calculations is BV/GE designs responsibilities. As soon as, these calculations are

finalized and approved, the calculated KVA/MVA rating of these transformers will be reflected in the PSAR accordingly.

2. Stuck breaker, defined as a breaker that fails to open under fault conditions, in a breaker and a half scheme will either result in tripping one adjacent line breaker or several bus breakers through breaker failure protection scheme.
   Opening of an adjacent line breaker will result into a second 345kv circuit being deenergized.
   Opening of several bus breakers will result into one 345kv bus being deenergized leaving all other 345kv circuits connected through the second bus without interruption.
   In conclusion, for a condition of unresponding "stuck" breaker, no more than one connection to either one transmission line or one main transformer will be affected.

3. This multiple type (EM) of relay features in the PSAR will be microprocessor based relay/EM.
   The protection system for main generator will be one set of conventional single function relays (solid state and/or EM) and duplicated by a microprocessor based digital relay system.

4. These generic transformers of protective relaying systems will be added in the PSAR.

5. Air break switch (ABS) feature as specified will be added.

6. Yes, the switchyard type (indoor) will be indicated in the PSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 08-006

PSAR Sections: 8.2 Offsite Power System

Question Date: December 15,1997

PSAR Question:

It is defined in ABWR SSAR 8.2.1.3 Separation:

(1) The reserve auxiliary transformer is separated from the main power and unit auxiliary transformers by a minimum distance of 15.24 m minimum separation be maintained between the switching stations and the incoming tie lines.

(2) The transformers are provided with oil collection pits and drains to a safe disposal area.

But there is no such a regulation defined in PSAR Sec. 8.2.1.3 Separation, please specify whether it is necessary to define this.

PSAR Response:

1. Lungmen project design is in compliance with GDC-17 regarding the physical separation requirements between the two offsite power supplies (345kV and 161kV).

   This is described in PSAR Sec. 8.2.2.1 which follows the applicable sections of ABWR SSAR Sec. 8.2.1.3 as follows:

   A. Physical separation distance between the main transformer bank and the reserve transformers is well in excess of the minimum requirement of 15.24 meters.

   B. The physical separation between the 345 and 161 kV switchyard enclosures is in excess of the 15.24 meters.

   C. The 345 kV and 161 kV circuits between the switchyard and the transformer area do not use overhead tie lines, but are installed in physically separated underground tunnels/trenches in compliance with IEEE Standard 765.

2. Lungmen project design is in compliance with GDC-17 requirement for separation oil collection pits around the power transformers, and drains to safe disposal area.

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

ROCAEC Review comments

(1) We agree to your responses in items 1.A and 1.B. For item 1.C, please describe in the PSAR the following: "switchyard to transformer is underground tunnels/trenches".

(2) Please describe in the PSAR the following: "The transformers are provided with oil collection pits and drains to a safe disposal area" (because it was not specified in the GDC 17).

Further Clarifications:

1. We will comply. It will be incorporated appropriately into the next PSAR revision submittal.

2. We will also comply. It will be incorporated appropriately into the next PSAR revision submittal.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 08-007

PSAR Sections: 8.3.1.1 Class 1E AC Power Distribution System

Question Date: March 20, 1998

PSAR Question:

Sections 8.3.1.1.1 and 8.3.1.1.7 stated that only when Class 1E degraded voltage is $\leq$70% of the bus voltage will it be switched to diesel generator. But in Section 8.3.1.1.5, it was stated that Class 1E electrical equipment will allow a voltage fluctuation of $\pm$ 10% of normal operating voltage (the starting transient voltage can go down to 70%). So, if the setpoint is at 70% for the switchover, could this low voltage cause damage to the equipment ? and how to avoid the damage ?

Response:

The equipment which will be powered from the Class 1E power system is specified to operate continuously with a voltage fluctuation of $\pm$10% of normal operating voltage and to operate momentarily with a voltage down to 70% of normal. Therefore this equipment will be designed to work within the design operating voltage range of the plant auxiliary power system without damage. If a Class 1E bus experiences degraded voltage to the 70% level, an alarm will be initiated, the bus will be tripped, and the EDG associated with that bus will start and connect to the bus. The loads will then be sequenced on to the bus. Therefore the equipment will be protected from exposure to sustained undervoltage.

ROCAEC's Review Comments:

Three or four years ago, ROCAEC contracted NUS to investigate the electrical capabilities of Chinshan NPS and raised the same question as 08-007. TPC should reference the resolution of Chinshan NPS to the question and supplement the response for Lungmen 08-007.

Further Clarification:

The responses is modified as follows:

The equipment which will be powered from the Class 1E power system is specified to operate continuously with a voltage fluctuation of ±10% of normal operating voltage and to operate momentarily with a voltage down to 70% of normal. As discussed in 8.3.1.1.7(8), if bus voltage degrades to 90% (or below) of its rated value, and after a time delay (to prevent triggering by transients), undervoltage will be annunciated in the control room. If the operator cannot correct the undervoltage within a given time, a fast transfer will occur, or the diesel generator will start and pick up the loads if the fast transfer source is not available. If a Class 1E bus experiences degraded voltage to the 70% level, an alarm will be initiated, the bus will be tripped, and the EDG associated with that bus will start and connect to the bus. The loads will then be sequenced on to the bus. Therefore the equipment will be protected from exposure to sustained undervoltage.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 08-008

PSAR Sections: 8.3.3

Question Date: May 15, 1998

PSAR Question:

Since Ni-H battery does not have the memory problem or the pollution problem as the lead battery which makes its maintenance a lot easier. Would Lungmen consider using Ni-H battery in place of the lead battery ? Please explain.

Response:

This topic has been discussed with battery manufacturers and it appears that there are no currently available station batteries available in the Ni-H (nickel metal hydride) design, nor are there any Class 1E batteries available in the Ni-H design. This is a relatively new technology which does not lend itself well to station battery design. The Lungmen batteries are a flat plate design which have been used and tested extensively over many years and have demonstrated no memory problem. According to the manufacturers surveyed, memory is a characteristic typical of wound designs, rather than flat plate designs. It is believed that the battery design utilized for the Lungmen plant have greater immunity to memory characteristic than Ni-H would have.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **N-09-001**

PSAR Sections: **Ch 9.2.11.3**

Question Date: **December 6, 1997**

PSAR Question:

1. In page 9.2-23, 4th line of this section stated that "The design pressure and temperature of RBCW equipment and piping are 1.034 MPaG and 171 ℃ maximum" which is different from Table 9.2-2d which listed the RBCW System Components' Design Temperature at 48.9 ℃. Please clarify.

2. The above pressure and temperature are different from the GESSAR (SSAR listed 1.37 MPaG and 70 ℃, respectively). Please explain the basic design difference of the RBCW system between Lungmen and GE ABWR.

Response:

1. The statement, "T*he design pressure and temperature of RBCW equipment and piping are 1.034 MPaG and 171 ℃ maximum"* is intended to identify that these are the **maximum** design conditions that can be found within the system, and are **not** the design conditions for the entire system. There are portions of the system which have design conditions lower than this. In fact, the majority of the piping and equipment within RBCW is designed for the lower temperature of 48.9 ℃. Only those parts of the system physically located within primary containment have a design temperature of 171 ℃.

   The equipment items identified in Table 9.2-2d are all located outside primary containment. This part of RBCW has a design temperature of 48.9 ℃.

2. The current design temperature of 48.9 ℃ is based on preliminary information, specifically, estimated water temperatures for the hot side of the various heat exchangers cooled by RBCW. When actual equipment is purchased, and detailed information is known, the design temperature will be re-evaluated. The design temperature value stated in the US ABWR SSAR was a preliminary estimate, based on previous experience for similar systems in similar plants. Final design conditions will be reflected in the FSAR.

The current design pressure 1.034 MPaG is based on detailed calculations modeling the pipe sizes and routing, and required flow rates, and expected pump curves, with margin. When the RBCW pumps are purchased, and detailed information is known, the design pressure will be re-evaluated. The design pressure value stated in the US ABWR SSAR was a preliminary estimate, based on previous experience for similar systems in similar plants. Final design conditions will be reflected in the FSAR.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-09-002

PSAR Sections: Ch 9.2.5, 9.2.11, 9.2.14, 9.2.15, 9.2.16

Question Date: December 6, 1997

PSAR Question:

Sections 9.2.5, 9.2.11, 9.2.14, 9.2.15 and 9.2.16 listed maximum sea water temperature of 35 ℃ and 34.7 ℃ as Design Bases. Is this temperature limit consistent with the highest temperature of Taiwan during summer time ? Please explain.

Response:

Information compiled to date regarding the local sea water temperature for the Lungmen site indicates that the maximum measured and recorded historical sea water temperature is 33.8 ℃. The basis for the maximum RBSW temperature of 35 ℃ is that it is a conservative value which envelopes the maximum sea water temperature of 33.8 ℃. Since TBSW is a non-safety system, a less conservative value of 34.7 ℃ is chosen as max. service water inlet temperature.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-003

PSAR Sections: Ch 9.1.2.1.3

Question Date: December 9, 1997

PSAR Question:

It was stated in Section 9.1.2.1.3 that the seismic category for the spent fuel pool liner will be listed in Table 3.2-1 but such information is not found in Table 3.2.-1. According to SRP, if liner is found not to be seismic category 1 equipment the additional evaluation should be performed. Please clarify.

Response:

The pool liners are being designed as Seismic Category I. The pool liners are considered to be part of the Auxiliary Fuel Building structure hence they fall into the same seismic classification as the building. The building is Seismic Category 1 as listed in Table 3.2-1

Further Clarification to ROC-AEC's Comments:

Table 3.2-1 for both the Reactor Building (U71) and the Auxiliary Fuel Building (U97) will be modified to include the "Spent Fuel Pool Liner", with a Safety Class of "3", Location of "SC" or "A" respectively, Quality Assurance of "S" and a Seismic Category of "I".

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-09-004

PSAR Sections: Ch 9.1.3.2

Question Date: December 16, 1997

PSAR Question:

1. In Section 9.1.3.2.2, 2nd paragraph, it was mentioned that "each of the two heat exchangers is designed to transfer one half of the system design heat load". Please clarify that the Auxiliary Fuel Pool Cooling and Cleanup System can meet the requirement that the water temperature be kept below 60 ℃ when single active failure is considered in SRP.

2. This section has not explained if the Auxiliary Fuel Pool Cooling and Cleanup System is designed for Seismic Class 1 ? According to the SRP 9.1.3 requirement on Spent Fuel Pool Cooling and Cleanup System, if the cooling portion is not designed for Seismic Class 1, then the makeup water system and building ventilation/filtration system must be designed for Seismic Class 1. Please clarify if this requirement is met.

3. Please explain if the Auxiliary Fuel Pool Makeup Water System complies with the SRP requirement that a Seismic Class 1 makeup water system and another reserve makeup water source are available ? Also, please explain where the water source is coming from and if it is Seismic Class 1.

Response: **(Revised per ROCAEC Comments regarding use of 10CFR50, SRP and ANS 57.2)**

General Response:

The Auxiliary Spent Fuel Pool Cooling and Cleanup (AFPC) system design, like the Reactor Building Spent Fuel Pool Cooling and Cleanup (FPCU) system design, meets the requirements of 10CFR50, SRP 9.1.3 and ANSI/ANS 57.2. However, with the small amount of decay heat generated from the old spent fuel there is much less urgency to restore normal cooling since the pool will heat up slowly. The AFP has been designed so that in the event that all normal cooling is lost, there is sufficient water in the pool to provide cooling and radiation shielding for a minimum of 30 days with no makeup water from outside sources required. During this 30 days period reasonable activities to restore normal cooling or to establish a means for adding water to the pool can be performed.

Response to part 1: To be conservative with heat removal capability and retain more flexibility in operating AFPC system, GE will design and supply an AFPC system with sufficient reserve margin such that with one AFPC heat exchanger removed from service the AFPC system will be able to meet the following conditions:

a) Capable of removing 100% of the heat load from the AFB pool maximum spent fuel storage design condition.

b) Capable of maintaining the AFB pool at a temperature below 140 °F.

Also the capabilities of one train of AFP cooling will be evaluated and the results will be included in the FSAR.

Response to part 2: As stated in the general response, the AFP has been designed so that in the event on all normal cooling is lost, there is sufficient water in the pool to provide cooling and radiation shielding for a minimum of 30 days with no makeup water from outside source required. Since the AFP is a seismic I structure, the reserved huge amount of water in AFP provides more inventory and inherently more reliable cooling water than any other feasible seismic category 1 makeup water supply system. This design satisfies the intent of providing a seismic category 1 makeup water system as required in SRP 9.1.3. In addition, portions of the AFB ventilation system that limit the potential for off-site radioactive contamination are designed to seismic category I requirements.

Response to part 3: As explained in above response to part 2, the 30 days passive capacity of the reserved water to cool and shield the fuel offsets the need for a seismic category 1 makeup water system. However, the AFP is provided with two outside makeup water sources:

1. Condensate Storage and Transfer System (CSTF) which provides normal makeup water to replace evaporative and leakage losses during normal operation and is not a seismic category I design as described in PSAR Section 9.1.3.2.2, and

2. Fire hoses which consist of the fire protection standpipes and hose connection in the AFB and the fire protection water supply (main yard, one diesel engine driven pump and water source) will be the seismic category I emergency back-up makeup water source.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: N-09-005

PSAR Sections: Ch 9.2.13

Question Date: December 19, 1997

PSAR Question:

In PSAR Section 9.2.13, the heat loads (Table 9.2-10) listed for Emergency Chilled Water System are quite different from the Heat Loads (Table 9.2-9) listed in GESSAR (Lungmen PSAR Heat Loads are smaller by over 100%). Please explain why the difference.

Response:

PSAR Table 9.2-10 was revised to reflect the Heat Load Adjustment Calculations performed after the SSAR. These calculations did not include the envelope loads or outside air loads. The Lungmen calculations are being finalized and will be included in the FSAR.

The PSAR will be revised to provide the SSAR values in the PSAR Table 9.2-10 as indicated in the attached table.

Table 9.2-10 Emergency Chilled Water System Heat Loads

Division	System	Normal Heat Load (kW)	Normal Chilled Water Flow (L/s)	Emergency Heat Load (kW)	Emergency Chilled Water Flow (L/s)
A	Reactor Building Electrical Equipment Area (A)	245	3.97	245	3.97
	Control Building Electrical Equipment Area (A)	350	5.61	350	5.61
	Total	595	9.58	595	9.58
B	Control Room Habitability Area	394	7.22	361	6.67
	Reactor Building Electrical Equipment Area (B)	256	4.17	256	4.17
	Control Building Electrical Equipment Area (B)	350	5.61	350	5.61
	Total	1,000	17.00	967	16.45
C	Control Room Habitability Area	394	7.22	361	6.67
	Reactor Building Electrical Equipment Area (C)	256	4.17	256	4.17
	Control Building Electrical Equipment Area (C)	350	5.61	350	5.61
	Total	1,000	17.00	967	16.45

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-006

PSAR Sections: Ch 9.1.4 and 9.1.2.2.1

Question Date: December 16, 1997

PSAR Question:

In SSAR, the Refueling Machine is classified to be Seismic Category 1 but in Lungmen PSAR it is modified to be Seismic Category IIA. Please explain the reason and the difference.

1. Because of the modification mentioned above, is the statement in Section 9.1.2.2 item (2) which says "There are no non-seismic systems,...located in the vicinity of the spent fuel pool or cask loading area on the refueling floor" still correct ? Please explain.
2. According to SRP 9.1.2 for spent fuel storage, consideration must be given to failures of any non-safety related system or non-seismic category 1 system around the fuel pool area that might cause Keff to increase over limit. Since Refueling machine has been downgraded to non-seismic category 1, any consideration has been give to this condition ? Please explain.

Response:

The refueling machine was classified IIA consistent with ANSI 57.1, the URD, as well as the majority of other equipment of a similar nature (such as the Fuel Preparation Machine or the Overhead Building Crane). There is no lessening of engineering requirements. The Refueling Machine must still be analyzed for dynamic events (SSE) and the same acceptance requirements apply.

1. The statement is un-changed as the Refueling Machine is no different than other equipment located in the vicinity and due to design features and analysis can not topple into the pool.

2. Yes, consideration has been given to this requirement, and it is felt the equipment is in compliance with the requirement. The requirements

is to evaluate all equipment to assure no equipment or devices failure may downgrade a Seismic Category I component. Per the requirements of this project, any device or piece of equipment which has the potential to downgrade a Seismic Category I component must be evaluated as a Seismic Category IIA item. The Refueling Machine shall be evaluated as a Seismic Category IIA piece of equipment.

__Further Clarification to ROC-AEC Comments:__

In the ABWR SSAR, Paragraph 3.2.5.1 (10) and Table 3.2-1 F5, the refueling platform is classified as non-safety related. However as noted, the refueling platform is of significant size and is located near essential components and thus consideration must be given to the consequences of failure. Thus in the event of an SSE, the machine must not fail and degrade an essential component. In the ABWR SSAR there are only two seismic categories, equipment must either be Seismic Category I or Non-seismic. To be consistent within the ABWR SSAR, the refueling machine was conservatively specified as Seismic Category I in order to meet the requirement that in the event of an SSE the machine would not fail and degrade an essential component. With the creation of the Utility Requirements Document (URD) which specifies Seismic Category II, the refueling machine can be correctly specified as Seismic Category IIA. That is, equipment which is not essential to safety, but is required to remain intact and not degrade essential components.

Additionally, the bid specification Appendix A, Paragraph 2.3.4.1.1 states .."Designs for the fuel handling system and equipment shall conform to ANSI/ANS-57. 1, Design Requirements for Light Water Reactor Fuel Handling Systems, except for the seismic classification of the BWR refueling platform assembly. The BWR refueling platform assembly shall be classified as Seismic Category II. Other exceptions to ANSI/ANS-57. 1 shall be specifically justified by the designer." This reflects the incorporation of the URD into the Bid Specification.

In consideration of the above, the refueling machine is more correctly defined as Seismic Category IIA for the Lungmen project. In addition to specifying the refueling machine as Seismic Category IIA, the first three sentences of subsection 3.9.1.4.7 should be revised as follows to eliminate any further ambiguities:

3.9.1.4.7

Storage, refueling, and servicing equipment which is important to safety is classified as essential components per the requirements of 10CFR50, Appendix A. This equipment is classified as seismic category I. Other non-safety related equipment which in the case of a failure, would degrade an essential component is defined in Section 9.1 and is classified as Seismic Category IIA. The Seismic Category I components are subject to an elastic dynamic finite element analysis to generate loadings.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **09-007**

PSAR Sections: **9.1**

Question Date: **December 2, 1997**

PSAR Question:

Is there any design difference of spent fuel pools for RB and AFB on aspects of thermal-hydraulic and structural? are the coolant make up system same for these two storage.

PSAR Response:

In general from a thermal-hydraulic stand point the heat load for AFPC system is lower than FPC due to age of the fuel being stored in the AFP. The make-up water source for AFPC system surge tank is the Condensate Storage and Transfer System (CSTF). The make-up water for FPC system is also replenished with CSTF, but during a seismic event, RHR or SPCU provide a Seismic Category I backup source of make up water from suppression pool to FPC. The latter is not applicable to AFPC. From structural stand point, although both AFPC and FPC pools are Seismic Category I, they differ in design load due to differences in temperature, elevation and hydrodynamic loads.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-008

PSAR Sections: Ch 9.2.5

Question Date: November 25, 1997

PSAR Question:

1. According to ABWR Standard SAR, the maximum UHS temperature is set to be 37.8°C as a safety design base. Why does PSAR set the maximum temperature (35°C) as a power generation design base. Please clarify the difference between these two design bases. In addition, how do you get the value of 35°C? What is the impact if the maximum UHS exceeds 35°C, please evaluate and provide us with your evaluation.

2. In Section 9.2.5.1(5), shall the non-Seismic Category S be read as non-Seismic Category I? Please clarify.

PSAR Response:

1. Information completed to date regarding the local sea water temperature for the Lungmen site indicates that the maximum measured and recorded historical sea water temperature is 33.8^{0}C. The basis for the maximum UHS temperature of 35^{0}C is a conservative value which envelopes the maximum sea water temperature of 33.8^{0}C.

   The ABWR RBSW ultimate heat sink (UHS) was based on the use of a spray pond which is not applicable for the Lungmen project.

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

2. Non-Seismic Category S is the mistake of Non-Seismic Category I.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-009

PSAR Sections: Ch 9.2.8.3

Question Date: November 24, 1997

PSAR Question:

In Section 9.2.8.3, it is mentioned that "The pretreatment system of the Condensate Makeup Purification System is chlorinated using a sodium hypochlorite at raw water intake". ROC-AEC would like to know what method to produce the sodium hypochlorite. Is this directly added by chemicals or by other methods. Please describe in detail.

PSAR Response:

The sodium hypochlorite will be purchased as a bulk commodity and dispensed , as required , by the dosing system of the Condensate Makeup Purification System.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-010

PSAR Sections: Ch 9.2.8.4.1

Question Date: November 24, 1997

PSAR Question:

In Section 9.2.8.4.1, it is mentioned that "When the effluent quality of a demineralizer becomes unsatisfactory, it is automatically removed from operation and the standby demineralizer is automatically put into operation". Please clarify the detection of effluent water chemistry is done by sampling system or on-line monitoring instrument. In addition, please describe what kind of monitor will be adopted, and also describe the specification of effluent chemistry.

PSAR Response:

The detection of effluent water chemistry is done by on-line monitoring instrumentation. On-line monitoring instrumentation associated with the condensate makeup demineralizer system is briefly summarized below.

Cation vessel effluent: Continuously monitor Sodium and pH

Anion vessel effluent: Continuously monitor chloride, silica, pH and conductivity.

Mixed Bed Vessel: Continuously monitor sodium, chloride, silica and conductivity.

Action level for regeneration of the vessels will be conducted on high sodium for the cation vessels, high silica for the anion vessels, and high sodium or silica on the mixed bed vessels.

Effluent chemistry specification for the demineralized water system effluent is consistent with the criteria established by the NI supplier in his "Plant Working Fluids Requirement Document" as provided below.

Parameter	Operating Target	Design Limit	Maximum Value
Chloride, PPB as Cl	2.5	4.0	25.0
Sulfate, PPB as SO_4	2.5	4.0	25.0
Conductivity, μS/cm	0.080	0.095	1.0
pH, pH units	6.7 to 7.3	6.5 to 7.5	6.0 to 8.0
Silica, PPB as SiO_2	5.0	10.0	50.0
Dissolved Oxygen, PPB as O_2	5.0	10.0	20.0
Corrosion Product Metals			
Iron, PPB as Fe	8.0	8.0	80.0
Copper, PPB as Cu	1.0	1.0	10.0
All Other Metals, PPB as the metal	1.0	1.0	10.0
Total Metals, PPB as the metal	10.0	10.0	100.0
Organic Impurities			
[Equivalent Δk,μS/cm)	0.2	0.4	2.0

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-011

PSAR Sections: Ch 9.2.9

Question Date: November 25, 1997

PSAR Question:

1. In Section 9.2.9.1 (2), the conductivity requirement for the Condensate Storage and Transfer System is set to be < 0.3 μS/cm at 25°C. Why is this requirement less conservative than those in ABWR SAR. Please clarify and explain.

2. In Table 9.2-6, minimum usable volume requirements for Condensate Storage Tank CST, what is Stone & Webster design basis? Have you ever performed any safety evaluation analysis to support those data in Table 9.2-6. In addition, does these data include the 570 m^3 of water for station blackout use to comply with RG 1.115?

3. Although the CSTF is defined as non-safety related in Section 9.2.9.3, can the Condensate Storage Tank remain available and/or operable during emergency operation. Please also refer to Section 9.2.9.1 (1) and provide your justification.

PSAR Response:

1. The conceptual design phase CST tank chemistry requirements were taken directly from TPC bid specification Table 1.5-3. As part of detailed design these values will be reconciled with the NI Supplier "Plant Working Fluid Requirements Document". .
   The conductivity listed in PSAR (<0.3μS/cm) is more conservative than that listed in ABWR SAR (<0.5μS/cm).

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

2. The usable volume of the Condensate Storage Tank is based on S&W standards of providing 135 percent of the condenser hotwell volume to

the high level point including system surge plus the required reactor reserve.

From TPC Bid Specification, the condenser hotwell is designed to store at the normal operating water level an amount of condensate equivalent to at least 4 minutes of full load operating flow. In addition, the hotwell is designed to have an available standby surge storage capacity (i.e. maximum water level) equivalent to 2 minutes of normal full load condensate flow.

Hotwell volume is calculated using Valves Wide Open flows. The volume required for reactor reserve is that volume required to supply the reactor for 8 hours during the Station Blackout (SBO) event. The reactor reserve volume can only be accessed by NI piping connections. All BOP connections are located on the CST above the water level representing the reactor reserve volume.

The volume of the Condensate Storage Tank allows the entire volume of water in the condenser hotwell to be transferred to the condensate storage tank when maintenance is being performed on the condenser and provides a volume to handle surges which may occur during normal plant operation. The Condensate Storage Tank is also capable of transferring 500 m^3 of water to the reactor well during refueling and also provides makeup water to the spent fuel pool to replenish the evaporation and leakage losses.

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

3. The tank was designed to meet all of the Nuclear Island requirements. The tank currently has no safety-related functions and is not seismically designed. The CST is enclosed in a weather protected dike designed in accordance with Regulatory Guide 1.143 because it contents are considered radioactively contaminated. The tank design also meets the requirements of NUREG 0800 Standard Review Plan 9.2.6 "Condensate Storage Facilities".

   The tank design also meets the requirements of NUREG 0800 Standard Review Plan 9.2.6 "Condensate Storage Facilities" will be added to the PSAR page 9.2-15.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-012

PSAR Sections: Ch 9.2.11

Question Date: November 25, 1997

PSAR Question:

GE ABWR Standard SAR, Chapter 9.2.11, the Power Generation Design Bases have included descriptions for "Loss of Preferred Power," whereas Lungmen does not have any discussions on the subject. Please clarify.

PSAR Response:

The TPC Bid Specification documents use the term "Loss of Offsite Power" (LOOP), which has been substituted in the Lungmen PSAR for the term "Loss of Preferred Power" (LOPP) as used in the US ABWR Standard Safety Analysis Report (US ABWR SSAR). These terms have the same meaning. As such, substituting LOOP for LOPP does not change the design of RBCW.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-013

PSAR Sections: Ch 9.2.12.4

Question Date: November 25, 1997

PSAR Question:

9.2.12.4 does not include inspection and test provisions for containment penetrations and associated isolation valves. Please explain.

PSAR Response:

Inspection and test provisions for containment isolation valves and piping for all systems are described in PSAR Section 6.6.

No change to the PSAR is required.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-014

PSAR Sections: Ch. 9.2.13

Question Date: November 25, 1997

PSAR Question:

1. Please explain any measures taken in system design to prevent/protect against water hammer during prompt restart of system.
2. Please clarify how to prevent "cycling on and off" when ECW pumps are powered from the emergency power.

PSAR Response:

1. The measures used to prevent water hammer include elevated surge tanks, high point vents, and slow acting valves.

2. If normal power is lost, the system is shut down until emergency power is provided. The pumps should be completely stopped when emergency power is provided. Thereafter, there should be no cycling on and off. Since this is a relatively rare occurrence, it will not be detrimental to system operation.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-015

PSAR Sections: Ch. 9.2 Chiller System

Question Date: November 25, 1997

PSAR Question:

1. Have environmental regulations been considered when choosing a refrigerant? What refrigerant will Lungmen use? Is there any possibility of changing out the refrigerant later on, due to environmental concerns?
2. If there is a need to change out refrigerant, is reduced capacity a problem?

PSAR Response:

1. Yes, refrigerant R-134a was chosen to comply with TPC Bid Specification Section 3.4.4.3.3 which requires the refrigerant used not be banned before the end of the chiller life. Refrigerant R-134a is not currently subject to a ban on its use. Since R-134a does not face a phase out or ban, manufacturers have not addressed a replacement refrigerant to be used in R-134a chillers. R-134a does not contain chlorine and does not contribute to ozone depletion. The chiller rooms and related ventilation systems will be designed in accordance with applicable codes such as ASHRAE 15 and NFPA 70, (ANSI B9.1 has been withdrawn and was incorporated into ASHRAE 15) and as required by TPC Bid Specification Section 3.4.4.3.3 . For both the ECW and NCW systems, refrigerant leak detectors will be provided in the chiller rooms with an alarm to the main control room upon detection of a refrigerant leak.

No change to the PSAR is required.

2. As indicated to the response to question 1, manufacturers have not addressed this concern since R-134a is not subject to phase out.

No change to the PSAR is required.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-016

PSAR Sections: Ch 9.2.15.2

Question Date: November 13, 1997

PSAR Question:

In page 9.2-35, it was mentioned that "Adequate corrosion and erosion safety factors shall be used to assure the integrity of the system during the life of the plant". Please clarify how adequate is adequate for the corrosion and erosion safety factors.

PSAR Response:

A corrosion/erosion allowance of 0.1 mm over the 40 year plant life has been utilized as a design basis of the RBSW 6% Molly Steel Pipe. This allowance is based on a maximum corrosion/erosion rate of less than 0.1 mill/year for the 6 % Molly Steel Pipe which has been utilized in sea water environments similar to that of Lungman. This piping is specified by project requirements as standard wall which for a typical 700 mm RBSW header will be 9.525 mm thick. Based on the design conditions, the required wall without a corrosion allowance is approximately 2.54 mm leaving approximately 6.985 mm additional wall for corrosion over the life of the plant.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-017

PSAR Sections: Ch 9.2.15 and 16

Question Date: November 13, 1997

PSAR Question:

There are not any descriptions in Section 9.2.15 and 9.2.16 for removing and reducing the effect of mud, silt, or organisms which are committed in PSAR Chapter 1, page 1C-64.

PSAR Response:

Please refer to Section 9.2.15.2 (Page 9.2-35), the first paragraph did state that BSW shall be able to function during abnormally high or low water levels and steps are taken to prevent organic fouling that may degrade system performance. These steps include trash racks, traveling water screens, electrochlorination treatment and automatic backwash strainers". Additionally, deposition and organic adhesion are precluded or minimized by maintaining water velocity in the RBSW piping above 3ft/sec* under all modes of operation and by the proper selection of piping material.

*: The purpose of stating the 3 fps is to say that if a velocity of greater than 3 fps is maintained, debris, silt, etc. will not settle out in the piping.

The pumphouse designs are based on Hydraulic Institute Standards which requires an approach velocity of no more than 1-2 ft/sec to minimize entertainment of silt into the pumphouse bays. Siltation of the intake structures is further mitigated by the deposition effect of the Intake Basin and the weir associated with the RBSW Submerged Reservoir.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-018

PSAR Sections: 9.3.2

Question Date: December 2, 1997

PSAR Question:

The PASS System in 2nd nuclear plant was designed and constructed by GE. Based on accident simulations conducted over the past years, the following deficiencies persist:

1. Are the gas samples adequately representative of the containment air?
2. Is there any packaged sampling and transporting system that will ease the sampling efforts such that the personnel involved would have the whole body dose < 0.05 Sv and the extremity dose < 0.5 Sv?
3. Is the high activity sample bottle easily removed from the lead container?
4. Sampling needles are easily bent and make the sampling difficult.

Response:

1. Yes. PASS shares the containment gas sampling line with CMS. Reference is T62 P&ID 31113-1T62-M2001 and M2005.

2. PASS is designed to meet NUREG-0737 as stated in Lungmen T62 SDD Sec. 3.4.3. NUREG-0737, Sec. II.B 3 (6) requires that radiation exposure to any individual to obtain and analyze a single sample of the reactor coolant and containment atmosphere shall be <0.05 Sv to the whole body and <0.75 Sv to the extremeties. The figures include sampling in the PASS room, and preparation and analysis (in the laboratory) of the sample.

3. GE drwgs 112D3678, 796E845 and 796E670 apply to the gas sample vial cask, small liquid volume cask and large liquid volume cask, respectively. The casks are designed to facilitate sample collection and removal for analysis. Once the procurement is made to the drawing, bottle removal is part of operating procedure, not purchase spec.

4. GE recommends maintaining 10 each spare hyperdermic needles. Luer-Lok Type BD, 25 gauge, 5/8" long, and Type BD, 18 gauge, 1" long.

Further Clarification to ROC-AEC's Comments:

Misalignment of PASS because of its tight tolerances, especially in the liquid tray, has been determined to be a primary contributor to the bending of sample needles. An inexperienced operator or a slightly misaligned liquid tray can result in the bending of needles.

In addition to maintaining a ready supply of replacements, operating plants have demonstrated successful use of the PASS by implementing design or operator enhancements to reduce the frequency of bent needles including: Applying marks on the floor beneath the tray, tape on the push-pull cable, and the use of locating pins inside the liquid tray to verify that the tray is in proper alignment.

Operating experiences from Taipower's existing nuclear power plants will be reviewed and applied to Lungmen PASS systems design whenever applicable.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-019

PSAR Sections: Ch.9.3.8

Question Date: November 24, 1997

PSAR Question:

In Section 9.3.8.2.2, it is mentioned that the " mixed waste is routed through oil separation/collection tank." What is the production rate for the separated waste oil? Where dose the oil transfer, and how to treat it?

PSAR Response:

The production rate for separated oil waste is approximately 880kg/yr/unit .This waste stream is transported to the Radwaste facility in 55gal drums and transferred into a waste oil tank sized for a three month station inventory. The contents of this nominal 500gal waste oil tank is used as feed stock for the radioactive waste incinerator.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-020

PSAR Sections: Ch. 9.5.1

Question Date: December 2, 1997

PSAR Question:

1. 9.5.1(1) states that the diesel fuel day tank is sufficient for usage of 8 hours, whereas PSAR indicates 5-8 hours. Please clarify.

2. 9.5.1(3) describes the use of the ROC FPC. Please clarify where the ROC FPC will be used. In general, US NFPA is more stringent than the ROC Fire Code. Why emphasize the ROC Fire Code?

Response:

1. We assume the question is to clarify the difference between the storage capacity for the diesel fuel day tank stated in the US ABWR SSAR, as compared to the Lungmen PSAR.

There are no regulatory requirements or regulatory guides which specify the total required storage capacity for integral or day tanks associated with standby diesel generator units. However, regarding minimum capacity requirements, US NRC Standard Review Plan, Section 9.5.4, Item III.6.e states:

"A low-level alarm is provided to enable the operator to accomplish minor repairs or maintenance before all fuel in the day or integral tank is consumed (assuming full-power operation)."

Also, ANSI N195/ANS 59.51, states:

"Each diesel shall be equipped with day or integral tank(s) whose capacity is sufficient to maintain at least 60 minutes of operation after reaching the low level alarm set point." Therefore, the sizing of the overall capacity of the day tank is at the discretion of the designer.

The above requirements, have been considered, in addition to TPC Bid

Specification, Appendix A, Chapter 11, Paragraph 11.5.10.5 which states:

"Each day tank shall have enough capacity to operate its associated standby power source for at least 4 hours at its maximum rated capacity and shall be designed so that when the level is reached where the fuel is automatically added, enough fuel remains in the tank to operate the unit for at least 60 minutes at its maximum rated capacity."

The interpretation of the above requirements is to provide a minimum storage capacity for each diesel fuel day tank of 5 hours.

Other reasons for the difference between the US ABWR SSAR, and the Lungmen PSAR include the diesel generator sizing criteria. The Lungmen diesel generators will be sized for a significantly higher power output than the US ABWR, which in turn requires a higher fuel consumption, and a physically larger diesel generator unit. These factors deem it necessary to minimize the size of ancillary equipment wherever possible, such as the day tank, since the larger diesel will be housed in a room of the Reactor Building which is the same size for Lungmen as it was for the US ABWR.

The actual capacity of the diesel fuel day tank(s) will be determined during detailed design, and recorded in the FSAR.

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

2. Use of ROC FPC hose streams as described in 9.5.1(3) and elsewhere in the PSAR reflect the flow requirements of TPC Bid Spec Section 3.4.6.1.6 for hose stations serving safety related equipment areas.

   The TPC Bid Spec Section 3.4.6.1.6 states:

   "A seismic category I piping system shall be designed by Contractor to supply water to dry standpipes and hose connections for manual fire fighting in areas containing equipments required for plant safe shutdown in the event of a safe shutdown earthquake. The piping system serving such hose stations shall be analyzed for SSE loading and shall be provided with supports to ensure integrity of system pressure boundary. The water supply system including tanks and pumps capable of providing flow (approximately 75 gpm per hose station) to at least two (2) hose stations (within NI buildings) will be provided by Others, with boundary at 1 meter outside the NI building walls."

A clarification of where the ROC FPC is used is contained in PSAR 9.5.1(6). This section indicates the general philosophy that will be used in the Fire Hazard Analysis to integrate requirements of the ROC FPC into the ABWR design. GE agrees that the intent of the CMEB 9.5-1 Fire Protection Program includes applicable safety-related areas as required by the Fire Hazards Analysis. The PSAR will be revised as follows:

Revised PSAR Section 9.5.1 (6) will state:

"...The Republic of China Fire Protection Code (ROC FPC) will be applied to the design of FP in all buildings and structures which are not required to perform safe shutdown. When possible, FP in these areas will also be designed to meet CMEB 9.5-1, 10CFR50 Appendix R, and NFPA; thereby providing a conservative design. However, when conflicts exist the ROC FPC will govern the design in these areas.

The BTP CMEB 9.5-1 and 10CFR50 Appendix R will govern the design of FP in all buildings and structures which are required to perform safe shutdown; and areas, systems, or components important to safety. The adequacy of fire protection for any particular plant safety system or area will be determined by analysis of the effects of the postulated fire relative to maintaining the ability to safely shut down the plant and minimize radioactive releases to the evironment in the event of fire. FP systems required by BTP CMEB 9.5-1 and 10CFR50 Appendix R will also be designed to meet applicable portions of the ROC FPC. However, when conflicts exist the BTP CMEB 9.5-1 and 10CFR50 will govern the design in these areas."

This description reflects the hierarchy of codes and standards required by TPC and documented in the Kickoff Meeting Minutes of November 20, 1996

Track Number: 09-020

PSAR Sections: Ch. 9.5.1

Question Date: December 2, 1997

PSAR Question #3:

Is 9.5.1.1.3(5) pointing to all HVAC ducts, or only meant for secondary containment ducts (6 of them)?

PSAR Response #3:

All NI HVAC ducts that penetrate a fire barrier are equipped with fire dampers of the appropriate fire rating. In general, fire barriers in buildings with equipment or functions required for safe shutdown (RB, CB, AFB) are rated for 3-hours.

At this time, the secondary containment ducts referenced are known to penetrate the 3-hour rated fire barriers between divisions. The intent of the ABWR HVAC design is to minimize the number of ducts which penetrate fire walls. When detailed design is completed, additional fire wall penetrations, if required, will be equipped with appropriate fire dampers, etc. and will be described in the FSAR.

Therefore, no changes to the PSAR will be made as a result of the response to the comment/question.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-020

PSAR Sections: Ch. 9.5.1

Question Date: December 2, 1997

PSAR Question #4:

9.5.1.1.6 does not explain why, in fire, the dual-damper will be switched from circulating system to once-through system. Please explain.

PSAR Response #4:

Section 9.5.1.1.6 is a general description of the smoke control features in the Lungmen plant. As indicated in Section 9.5.1.16, detailed descriptions of each HVAC systems smoke control features are provided in Sections 9.4.1 and 9.4.5. To clarify Section 9.5.1.1.6, a paragraph will be added to PSAR Section 9.5.1.1.6 immediately before the sentence, "For detailed description for RB and CB HVAC ....". The paragraph will read as follows

"For HVAC systems that recirculate air during normal operation, the return air damper will be closed during smoke removal mode and the system will be operated as a once through system to maximize effectiveness of the smoke removal mode."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-021

PSAR Sections: 9.5.2

Question Date: November 3, 1997

PSAR Question:

1. Will portable wireless transmitter/receivers arouse electromagnetic interference problems?

2. Is there a Redundancy considered in communication subsystem such as PABX, PA? Because some systems may need to automatically reset due to the unknown accumulated error code after running for a period of time.

PSAR Response:

1. Power levels and frequency bands of the portable wireless system are selected to eliminate noise interference or actuation on sensitive instrumentation and control systems.
2. Private Automatic Exchange (PAX) system, as well as Sound Powered Telephone System are employed as alternative (redundant) communication system to complement the Portable Wireless Telecommunication System (PWT).

ROCAEC Review Comments:

1. As described in PSAR Section 9.5.2, in the area where the portable wireless communication is unable to use, the replacement is listed in Section 9.5.13.14. However, this Section is not shown in the PSAR. Please explain.
2. Is there any nuclear power plant that use portable wireless communication system? Please provide this information.
3. TPC's microwave communication system was not addressed in PSAR page 9.5-31,32 for external communications. Is this portion covered? Could it be an editorial mistake that the Emergency Telephone including the communication with "NRC" ?

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Further Clarifications:

1. The reference to section 9.5.13.14 is an error. The sentence should be revised to read: "Where portable, wireless communications units can not be effectively used the primary alternate systems are the Intraplant (PABX) Telephone System for private communication and the Intraplant Public Address (PA) System for party line communication. In the event of loss of power the Sound-Powered Telephone and Portable UHF/VHF Radio systems may be implemented."

2. The following nuclear plants are known to us to have satisfactorily tested the use of portable wireless telephone systems in nuclear plants:
   a) Carolina Power and Light, H. B. Robinson Steam Electric Plant
   b) Consolidated Edison, Indian Point 2 Nuclear Power Station
   c) Boston Edison, Pilgrim Nuclear Power Station
   d) Northeast Utilities, Millstone

3. Yes. The first sentence of point (5), section 9.5.2.2 (page 9.5-31) will be revised to read as follows:
   "Normal offsite communications is provided by public telephone lines and the utility microwave radio system network which is connected to the PABX telephone switch."

   Section 9.5.2.2(5)(a) has an editorial error and should read:
   "(a) Emergency Notification System (ENS) - Provides a communications link with the ROC-AEC."

   Section 9.5.2.2(5)(b): Replace "NRC" with "ROC-AEC."

   PSAR will be revised as stated above.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-022

PSAR Sections: Ch. 9.5.3, 9.5.3.1.1

Question Date: December 2, 1997

PSAR Question:

Standby AC lighting system should be added to the lighting system. When normal AC lighting system is lost, this system will provide lighting for sufficient length of time for prolonged period of power outage.

1. Where are the non-class 1E battery packs and class 1E battery packs located?
2. Will the explosion-proof type be provided in the diesel generator room?

PSAR Response:

The non-Class 1E standby AC lighting system does not apply to the Lungmen plant as there is no back-up power source for this system. The functions of the standby AC lighting system have been considered in the Normal AC lighting subsystems. However, the emergency AC lighting system is backed up by the emergency diesel generators, and supplies alternate lighting to safety-related areas.

1. The battery packs are part of the guide lamp system. As discussed in PSAR section 9.5.3.2.3, the DC guide lamps provide lighting in safety-related areas such as the MCR, remote shutdown panel room, and in passageways, egress routes, and stairways throughout the plant. Egress routes and passageways from non-safety-related areas will have non-class 1E guide lamps.

2. The diesel generator room does not include an explosion-proof design. The systems which include explosion-proof fixtures are listed in 9.5.3.1.1(5)(l). The diesel generator rooms contain a significant amount of equipment which is not of an explosion-proof design. The ABWR SSAR included explosion proof lighting fixtures in the diesel generator rooms, however these rooms are not classified as hazardous areas and, as such, do not have a requirement for explosion proof lighting fixtures. A requirement to use explosion proof fixtures implies that everything in the room must be compatible with a hazardous environment. We believe that this is difficult, if not impossible, to achieve. The prime example is the generator, which is an air-cooled design. This would not be available in an explosion proof design. Based upon the definitions of hazardous areas in National Electric Code article 500, we consider that the diesel generator rooms are not hazardous areas and do not require explosion proof light fixtures.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-023

PSAR Sections: Ch. 9.5.3.2.2

Question Date: December 2, 1997

PSAR Question:

How long can the emergency DC lighting system last?

PSAR Response:

The DC battery calculations consider that the DC emergency lighting will remain illuminated for 2 hours during a loss of offsite power. This is considered adequate time to reestablish an AC power source for the safety related lighting, and to perform the tasks necessary in the areas supported by the system.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-024

PSAR Sections: 9.5.10

Question Date: December 2, 1997

PSAR Question:

1. The M-G Sets input voltage is 13.8 kV, but there is no mentioning of the output voltage.

2. If M-G Sets Output Voltage is 13.8 kV, then RIP voltage should be 13.8 kV. However, ABWR and K6/7 RIP uses 6.9 kV voltage. Does the RIP manufacturer supply to 13.8 kV specification? Please investigate.

PSAR Response :

1. The M-G Set induction motor receives its input from the associated 13.8 kV power bus and the design rated voltage of the induction motor is 13.2 kV at the induction motor terminals (lower voltage to account for voltage drop from power bus to the motor input terminals). The M-G Set generator provides the output power of the M-G Set and the output voltage rating is 13.8 kV, which is the same as the voltage rating of the power bus that provides input power to the M-G Set. This allows the voltage ratings of all of the RIP ASD Input Power Transformers to be the same (i.e. 13.8 kV power source feeds all 10 RIP ASD Input Power Transformers, regardless of whether the M-G Set Generator output is the power source, which applies for 6 RIP ASDs of each plant; OR a 13.8 kV power bus is the power source, which applies for the 4 RIP ASDs of each plant that do not receive input power from an M-G Set).

   No changes to the PSAR are required as a result of this question/response.

2. The RIP motors themselves never see 13.8 kV; rather, their voltage and frequency vary together (such that the volts/hz ratio is maintained

constant) in order to control the RIP speed. The range of voltage and frequency is controlled by the ASD, which feeds a maximum of approximately 3000V to its associated RIP to achieve rated speed. The RIP ASD input power transformers for Lungmen are specifically designed to accommodate the difference in the incoming voltage (i.e., 13.8 kV) from the 6.9 kV basis values of the K6/7 and US ABWR designs. Thus, the manufacturer can supply the RIP ASD and RIPs to the same voltage specifications as previously used.

No changes to the PSAR are required as a result of this question/response.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-025

PSAR Sections: 9.4

Question Date: December 2, 1997

PSAR Question:

1. The uncontaminated air from primary containment will be circulated to the secondary containment. In case where the primary containment air is containment air is contaminated, the air is processed through SGT and discharged. Suggest the radioactive set point of the primary containment contaminated air be incorporated into the PSAR.

2. The design temperature for the drywell cooling system (57oC to 93oC) is much higher than other areas of the plant (10 to 40oC). Why?

PSAR Response:

1. During outages when the primary containment requires personnel entry, air is exhausted from the primary containment to the Reactor Building secondary containment exhaust duct system and is exhausted to atmosphere. It does not circulate within the secondary containment. If the primary containment air is contaminated, the exhaust flow is directed to the SGT. The following sentence will be added to the PSAR, Section 9.4.5.6.2, second paragraph, after the second sentence:

   "The SGT filter units are located on the northeast quadrant of the Reactor Building at the 23500 elevation. SGT discharge is to a stack located on the Reactor Building roof. Setpoints associated with the SGT radiation monitors will be located in the Offsite Dose Calculation Manual (ODCM), and will be provided by the licensee at the FSAR submittal stage. Additional information pertaining to the ODCM can be found in sections 16.5.4.2.1 and 16.5.4.2.4."

2. The primary containment is inaccessible during normal plant operation due to high radiation levels and the inerting of the drywell with nitrogen. The equipment in the drywell is designed and qualified for the drywell

temperatures. During maintenance operations that require personnel access to the drywell, the temperature is maintained at a lower temperature (approximately 30°C and a relative humidity of 40 %). The temperatures in the drywell are consistent with previous boiling water reactor plants.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-026

PSAR Sections: 9.1

Question Date: December 2, 1997

PSAR Question #1:

How do you estimate the amount of spent fuel resulting from 40 years of operation? Based on this estimation, how do you estimate the space required for the fuel storage both for Reactor Building and Auxiliary Fuel Building?

PSAR Response #1: **(Revised per ROCAEC Comments regarding use of 10CFR50, SRP and ANS 57.2)**

The space of the spent fuel pool and the related clean-up and cooling system shall have the capability of accommodating unconsolidated spent fuel assemblies resulting from forty (40) calendar years of plant operation plus the total number of assemblies in one core, assuming that the reactor is operated at full power of 18-month cycles of plant operation and 87% availability. Spent fuel storage of 40 years reactor operation is to be provided by a) 15 years storage capacity in each Reactor Building spent fuel pool and b) a combined 50 years (2 x 25 years) spent fuel storage in an Auxiliary Spent Fuel Pool. The Auxiliary Spent Fuel Pool will be a wet storage pool shared by both units on the Lungmen site, housed in a facility which also includes the swing diesel generator common to both units. The Auxiliary Fuel Building including its pool structure meet the requirements of 10CFR50 and SRP 3.8.4.

The basis for the batch size are provided in the table below.

Parameter	**Value**
Thermal Power Rating	3926 MWt
Refueling Interval	18 months (1.5 years)
Cycle Length	431 EFPD
Capacity Factor	79%
Batch Average Discharge Exposure	42.5 GWD/MT
GE12 Design Mass	181 Kg Uranium

A reload size is calculated as follows:

$$\frac{(3926\ \text{MWt}) \times (431\ \text{EFPD/cycle}) \times (1\ \text{cycle})}{(42500\ \text{MWD/MT}) \times (0.181\ \text{MT/fuel bundle})} = 220 \text{ fuel bundles per cycle}$$

On site storage in the Reactor Building will be provided in the single spent fuel pool for each unit for 3072 unconsolidated spent fuel assemblies, which equals

(872 bundles for a full core) + (220 bundles/cycle x 10 cycles)

For spent fuel storage in the Auxiliary Spent Fuel Pool, the required storage capacity is calculated below:

220 bundles/cycle x 16 cycles x 2 units = 7040 bundles

The current design storage capacity of the Auxiliary Fuel Pool is 8760 fuel bundles which is greater than the required 7040 fuel bundles as calculated above. This extra capacity design is only for potential future storage. With the designed storage capacity, the Spent Fuel Pools in both Reactor Buildings together with the Auxiliary Spent Fuel Pool in Auxiliary Fuel Building can accept in the future up to 252 fuel bundles for each refueling cycle plus one full core of each unit.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-026

PSAR Sections: 9.1

Question Date: December 2, 1997

PSAR Question #2:

The new fuel and spent fuel storage racks are of the same high density design. Is the any design base difference between these racks on aspects of nuclear design, mechanical design, thermal-hydraulic design, material consideration, and dynamic analysis?

PSAR Response #2:

The fabricated assemblies are expected to be identical. Thus the mechanical design and material considerations will apply to the new and spent fuel storage racks. During analysis, spent fuel storage racks will consider the effects of water mass as part of the dynamic analysis, where as the new fuel storage are dry. The same is applicable for the thermo-hydraulic analysis. For the new fuel storage racks, an additional calculation shall be performed to consider the effects of aqueous foam, small droplets, spray, or fogging. The applicable standards for the new fuel storage will be ANSI 57.3 and for spent fuel storage the applicable standard will be ANSI 57.2.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-027

PSAR Sections: 9.5.1

Question Date: December 2, 1997

PSAR Question:

1. Under the worst condition for a single fire protection zone, will cable raceway fire affect the substitute shutdown capability ? Please explain.
2. It was explained in PSAR that fire detection system will not be installed for the switchbox of the control panels in the MCR as recommended by BTPCMEB 9.5-1 Sec. 7b. But whether the fire detection system is installed for the MCR roofing and surrounding rooms is not mentioned. Please clarify.

PSAR Response:

1. No. The ABWR design assumes that in the event of a fire in a divisional fire zone, the divisional equipement is immediately no longer available to provide its safety function. In most all single fire zones, the complete burnout of the fire zone can be tolerated without affecting the safe shutdown cabablily. This is because the fire barrier system assures that two safe shutdown divisions will be free of fire damage, when a fire occurs in the remaining division. This is described in PSAR section 9.5.1.1.

   For some fire zones, such as Primary Containment (Fire Zone F1900) and the MCR (Fire Zone FC4910), and other Special Cases, the function of these areas require them to contain equipment from all four safety divisions. For these zones, the ABWR design provides alternative means to ensure safe shutodown capability. These means are described in PSAR Sections 9.5.1.3.11, and analyzed in Appendix 9A.5.

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

2. Fire detection will be installed for the MCR at ceiling level, and surrounding rooms at ceiling level. These systems are provided as required by BTP CMEB 9.5.1.C.6.a(1); in accordance with PSAR Section 9.5.1 (6).

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-028

PSAR Sections: Ch 9.1.3

Question Date: January 5, 1998

PSAR Question:

The heat exchanger capacity for the fuel pool cooling and cleanup system in Lungmen PSAR was listed at 1.92MW (6.55 x 10^6BTU/hr) which is about half of 13.6 x 10^6BTU/hr used at the Second Nuclear Power Station . But Lungmen has higher output than the Second NPS. Is this reasonable ? Please also explain if the Lungmen fuel pool cooling and cleanup system can satisfy the SRP requirement that "under Single Active Failure and Max. Normal Heat Load conditions, water temperature will be maintained below 60^0C and under Max. Abnormal Heat Load condition, pool water will not boil over".

Response:

The calculated size (including design margin) of the FPCU heat exchangers is 2.0 MW (6.83 MBtu/hr). The 1.92 MW listed in Section 9.1.3 will be updated to 2.0 MW to be consistent with the current FPCU heat exchanger design. The heat removal capacity of each FPCU heat exchanger is based on removing 50% of the normal heat load based on the fuel pool gate being closed at 17 days after shutdown (CRD insertion).

The combined heat removal capacity of two FPCU heat exchangers plus one RHR heat exchanger is sufficient to remove 100% of the maximum (abnormal) heat load based on the fuel pool gate being closed at 120 hours (5 days) after shutdown.

The heat load conditions existing in the spent fuel pool are as defined below:

Normal Heat load

Normal heat load is the decay heat from the accumulation of 15 years of spent fuel with the newest spent fuel batch

(approximately 25% of the full core) having just been placed in the fuel pool 120 hours (5 days) after shutdown.

Maximum Heat Load

Maximum heat load is the decay heat from the accumulation of 15 years of spent fuel with the newest spent fuel batch of 100% of the full core having just been placed in the fuel pool 120 hours (5 days) after shutdown.

During the maximum heat load condition or during the normal heat load condition when the fuel gate is closed between 5 and 17 days after shutdown, a RHR heat exchanger is utilized to supplement FPCU in cooling the spent fuel pool. This is accomplished through the piping cross ties between FPCU and RHR which are designed to allow a flow of water (350 m^3/hr) from the FPCU skimmer surge tanks to be cooled by the RHR heat exchanger and returned to the spent fuel pool.

If a single active failure occurs within FPCU, the remaining FPCU heat exchanger, along with supplemental cooling from the RHR heat exchanger as needed, will maintain the spent fuel pool temperature as follows:

With the normal heat load in the spent fuel pool, but with a single active failure within FPCU, FPCU will maintain the spent fuel pool bulk water temperature below 60^{0}C (140^{0}F).

With the maximum heat load in the spent fuel pool, but with single active failure within FPCU, FPCU, in conjunction with supplementary cooling from RHR, will maintain the spent fuel pool bulk water temperature below 81^{0}C (180^{0}F).

The performance of the FPCU and RHR heat exchangers, as described above, will be confirmed in the Lungmen fuel pool cooling analysis which will be summarized in Table 9.1-12 to be included in the Lungmen FSAR. Based on the fuel pool cooling analysis performed for the SSAR (summarized in SSAR Table 9.1-12) and other ABWR designs, the heat capacity of the FPCU and RHR heat exchangers has been confirmed to be sufficient to maintain the spent fuel pool temperature below the temperature limits specified above.

Further Clarification to ROC-AEC's Comments(04/15/98)

The results of preliminary calculations, based on the current FPCU heat exchanger design, show that if RHR cooling is not available at 5 days after shutdown when the fuel gates are closed with the normal heat load in the fuel pool, the two FPCU heat exchangers can easily maintain the fuel pool temperature below the 60°C (140°F) limit specified in Section III.1d of SRP 9.1.3. The estimated peak temperature of the fuel pool water is 52°C (126°F) at approximately 6 days after the fuel gates are closed and continues to cool to below 49°C (120°F) at 13 days. Per Section III.1d of SRP 9.1.3, a single active failure does not need to be considered for maximum heat load conditions.

Based on the above, the FPCU heat exchangers are adequately designed. Table 9.1-12, to be included in the Lungmen FSAR, will summarize the results of the fuel pool cooling analysis including a case for single active failure in RHR.

ROCAEC Review Comments:

The response was based on 25% Full Core for each Refuel Batch Size to calculate the heat load. Please clarify the following questions :

(1) If 18 months is the refueling cycle, what is the largest refueling batch as a percentage of full core ? Is it higher than 25% ?

(2) According to the SDD 3.2.2.2 provided by TPC, Refueling Batch is 25 ~ 29% of full core. And in answering the 09-026 question, it was indicated that 252 fuel bundles (28.9% of full core) is possible since ABWR Capacity Factor is better. So the response that 25% was used for heat load analysis seems not conservative enough and larger number should be used ?

(3) According to the response to PSAR 09-026 question, refueling batch will reach equilibrium at cycle 5. What is the largest batch size as a percentage of full core before equilibrium ? Calculation of heat loads should consider the largest batch size before equilibrium as well.

(4) From the considerations discussed above on the refueling batch size, can the

requirement ≤ 60 °C still be met under Normal Heat Load (including operation of 2 strings of FPCU or one string of RHR plus one string of FPCU) ? If under abnormal heat load conditions, can relevent requirements be met too ?

(5) The previous response stated that the size of the FPCU heat exchanger is 2.0 MW. Please explain what temperature was used for the heat exchanger pool side exit to achieve this value ? If the pool side exit temperature is 60 °C, what is the heat removal capability of the heat exchanger ?

Further Clarifications:

(1) The largest refueling batch size has not been determined and will be dependent of how the plant is operated. However, the 29% of full core batch size (252 bundles), as specified in the response to 09-026, has been used as the basis for the fuel pool heat load calculation and is considered realistically as an upper limit based on a 90% capacity factor.

(2) The fuel pool normal heat load used in sizing the FPCU heat exchangers is based on the 29% of full core refueling batch size (252 bundles).

(3) See responses to 1 and 2 above. Adequate margin has been included in the FPCU heat exchanger sizing to account for uncertainties in the refueling batch size. The 25% mentioned in the response above was provided as a descriptive estimate of the batch size and not the upper limit 29% batch size which is being used in the calculations.

(4) Yes, per responses to 1 and 2 above.

(5) The 2.0 MW heat capacity of each FPCU heat exchanger is based on removing 50% of the normal heat load when the fuel pool gate is closed at 17 days after shutdown (CRD insertion). The FPCU heat exchangers are designed to remove the 2.0 MW of heat when the inlet fuel pool water temperature is at 49 °C (120 °F). The heat removal capacity of the FPCU heat exchangers will increase as the inlet fuel pool temperature increases. The actual performance of the FPCU heat exchangers will be provided in the Lungmen fuel pool cooling analysis which will be summarized in Table 9.1-12 to be included in the Lungmen FSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-029

PSAR Sections: 9.3.5

Question Date: February 23, 1998

PSAR Question:

1. The conventional Explosive Valves used for the injection valve of Standby Liquid Control System (SLCS) has been replaced by the Motor-Operated-Valves (MOV). Please explain the reason for this switch in design.

2. **The injection path of SLCS is through the HPCF system piping in the PSAR which is different from the conventional path of through the vertical injection piping under the lower core plate. Please evaluate the effectiveness and evenness of neutron poison after its injection into the reactor and whether it will safely shutdown the reactor during accident.**

Response:

1. The use of Motor-Operated-Valves (MOVs) instead of explosive valves for SLCS design has the following advantages:

   a) Piping design is simplified, thereby avoiding additional pipe protection design for thermal expansion of heated pipe and possible hydrodynamic loads during initiation of explosive valves.

   b) Maintenance service is reduced and periodic replacement of charges of explosive valves is not required. This also reduces radiation exposure.

   c) Post-injection containment isolation capability is enhanced without degrading system reliability. (MOVs can be closed following boron injection; explosive valves cannot be closed to provide containment isolation.)

2. The injection path of SLCS through the HPCF sparger above the reactor core has been shown to provide adequate mixing to meet system requirements by a testing program performed by GE. The objective of the test program was to confirm the acceptability of the boron mixing and that the reactor could be safely shutdown.

   A 1/6 scaled three-dimensional (3-D) model of a BWR reactor pressure vessel (RPV) with internal pumps was used in these tests to simulate, measure, and study the boron mixing characteristics of the boron solution. A tracer material, simulating the boron, was injected into the scale model of the BWR RPV and the mixing of the two liquids was observed to confirm the mixing phenomena. The time and volumetric mixing effects were monitored from the start of injection until such time as the concentration of the injected material reaches the level required to bring the reactor subcritical. The tests were performed over a wide parametric range (0-25%) of reactor core flow rate because of its primary influence on the mixing phenomena and thus enveloping the ATWS conditions of low core flow rate.

   Test data was analyzed and evaluated to confirm the applicability of the scaling laws so that the test data could be utilized to develop a boron mixing analytical model applicable to the advanced boiling water reactor (ABWR). The analytical model and test data were used for design/evaluation for boron injection and mixing to confirm the capability of the SLCS with an injection path through the HPCF to safely shutdown the reactor.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-030

PSAR Sections: 9.5.1, 9A, 9B

Question Date: March 2, 1998

PSAR Question:

1. Fire Hazards Review of Area with RBSW Equipment

   Per BTP CMEB 9.5-1, Position C.1.b, the fire hazards analysis should "...(2) determine the consequences of fire in any location in the plant on the ability to safety shut down the reactor or on the ability to minimize and control the release of radioactivity to the environment, and (3) specify measures of fire prevention, fire detection, fire suppression, and fire containment and alternative shutdown capability as required for each fire area containing structures, systems, and components important to safety that are in conformance with NRC guidelines and regulations." Contrary to this, no discussion is provided nor indications that the final FHA will address those areas containing components (including electrical circuits) for the Reactor Building Service Water (RBSW) system that are not located in the PSAR section 9A.2.5 identifies the RBSW as a core cooling system. Identify the applicable areas and describe the criteria to be used for :

   - Protection against fire (barriers, detection systems, and suppression systems),
   - Consequences of a fire (impact on safe shutdown, smoke removal, application of fire suppression), and
   - Consequences of inadvertent operation or rupture of fire suppression systems.

   Also, identify other systems outside the Reactor Building, Control Building, Turbine Building, and Radwaste Building that fall under this CMEB position.

PSAR Response:

The Fire Hazards Analysis (FHA) will be performed in accordance with the requirements of BTP CMEB 9.5-1 and the results reported in a separate report and summarized in the FSAR. The PSAR sections that discuss the FHA were included to provide a brief description of the methods used and a sample of the format and content of the final FHA.

The final FHA will address all the areas of concern as described in the PSAR including protection against fire, consequences of fire, and consequences of inadvertent operation or rupture of fire suppression systems.

Regarding the RBSW system, we have included below a brief summary of the RBSW Pump House that houses the RBSW pumps and associated equipment . The details of the final analysis will be described in the final FHA and summarized in the FSAR.

The RBSW system and components are located in the RBSW Pump House, the Control Building and associated pipe tunnels and trenches..

The RBSW Pump Structure is a shared facility for Unit 1 and Unit 2. The RBSW components for Unit 1 and Unit 2 are separated in accordance with the requirements of BTP CMEB 9.5.1.

The primary fire suppression in The RBSW Pump House is provided by portable fire extinguishers with hose rack stations as a backup. Fire detection in RBSW Pump House is accomplished by the central alarm system designed in accordance with NFPA 72.

Floor drainage system is provided in the pump cubicles to adequately control the inventory resulting from actuation of the fire suppression system and in case of a pipe rupture .

In addition to the Reactor Building, Turbine Building, Control Building and the Radwaste Building the Fire Hazards Review, at a minimum, in its response to BTD CMEB 9-5-1 Position c / b will address the Fire Pump house, Auxiliary Building, Fuel Building, Switchgear Building, Access Control Building, Switchyard and miscellaneous yard structures (including transformers, yard tankage and tunnels.).

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-030

PSAR Sections: 9A

Question Date: March 6, 1998

PSAR Question:

2. Fire Hazards Review Of Shared Facilities

Lungmen NPS is a two unit plant that has some facilities shared, e.g., the Radwaste Building and the Auxiliary Fuel Building, BTP CMEB 9.5-1, Position C.1.b requires that fires involving facilities shared between units and fires due to man-made site-related events that could affect more than one reactor unit (such as an aircraft crash) be considered. PSAR section 9A does not identify criteria or the approach to handling fires in these shared facilities in the FHA. A specific concern is the swing diesel generator installed in the auxiliary Fuel Building. Although this swing diesel has not been identified as a safe shutdown component, it has circuits associated with the electrical systems of both units. Describe the criteria and approach to be used in the formal FHA to evaluate the impact of these and any other shared facilities on the safe shutdown of both plants. Describe the criteria for the fire protection features for these facilities. Specifically address the swing diesel.

PSAR Response:

2. Aircraft crashes which is not a credible event for the Lungmen NPS. Please see section 2.2.3.5 Regarding the Auxiliary Fuel Building (AFB) and the swing emergency diesel generator (EDG) for the following responses to the criteria and approach to be used in the formal FHA to evaluate the impact of these and any other shared facilities on the safe shutdown of both plants:

- The Lungmen auxiliary power system uses three divisional diesel generators and one swing diesel generator to provide emergency power to safety related systems over the full spectrum of design basis events. The swing diesel generator is not normally required to mitigate design basis events, but it can for this purpose in place of a divisional diesel generator which is not available due to maintenance or other reasons.

- The swing diesel generator electrical configuration uses IEEE 384 separation methods to preclude the possibility of a common mode failure disabling more than one diesel generator during a fire in the Auxiliary Fuel Building. This ensures that at least two trains of electrical equipment are available to mitigate the consequence of a design basis accident or bring the operating units down to a hot or cold shutdown condition.
- No change will be made to the PSAR as a result of the response to this question. The above criteria and approach will be used in the formal FHA.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-030

PSAR Sections: 9.5.1, 9A, 9B

Question Date: March 2, 1998

PSAR Question:

3. Fire Hazards Review Of Construction Activities

   Lungmen is a two unit station. It is likely that one unit will be operating while the other is still under construction. BTP CMEB 9.5-1, Position C.1.e requires continuing evaluation of fire hazards during the construction. PSAR section 9a.2 does not identify any criteria regarding the identification of and protection from construction fire hazards. Discuss the criteria and approach to be used in the formal FHA to evaluate the impact of construction activities at the second unit on the FHA for the operating first unit. Include discussion of the configuration of the fire protection water supply system during the construction period.

PSAR Response:

A. The Yard Fire Main is composed of a loop around each unit with the common leg to be installed with unit 1 side of the temporary security fence required for completion of unit 2.
B. The Lungman Site Fire Protection Program will be in place and will address emergency action in case of fires on the construction site.
C. Part of the completion program for unit 1 will be an engineering review to verify that there are no temporary electrical or mechanical system cross connects to unit 2 which could invalidate the unit 1 Fire Hazards analysis or
   the safe shutdown analysis.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-030

PSAR Sections: 9A

Question Date: March 6, 1998

PSAR Question:

4. **Fire Hazards Review of External Hazards**

   BTP CMEB 9.5-1, Position C.1.b requires that fires due to man-made site-related events that could affect more than one reactor unit (such as an aircraft crash) be considered. Per PSAR section 9.5.1.1.6, the safety related buildings have exterior walls designated as 3 hour fire barriers to protect safe shutdown equipment from external fires. One hazard from external fires is smoke. According to PSAR section 9.5.1.1.6, fire dampers are omitted from HVAC supply ducts. As a result, smoke form the external fire may be drawn into the safety related buildings, hampering shutdown equipment and activities and possibly activating various smoke alarms. The control room HVAC design considers external smoke hazards, but other areas of the plant do not.

   Describe the criteria and approach to be used in the formal FHA to evaluate the impact of external hazards on safe shutdown, including the migration of smoke into areas containing safe shutdown equipment.

PSAR Response:

4. Regarding aircraft crashes, GE has demonstrated that an aircraft crash is not a credible event for the Lungmen NPS. Please see section 2.2.3.5.

   Regarding external smoke hazards, section 9.5.1.1.6 will be modified for the FSAR to include combination, operable smoke-fire dampers for external openings in fire rated sections of external walls except as discussed below. The dampers will be rated according to UL 555 and 555S. A fusible link in the actuator mechanism will close the damper in the presence of fire. Smoke detectors in the supply duct of HVAC systems will close the dampers and de-energize the fans according to

NFPA 90A, to prevent the movement of smoke into buildings.

In the Control Room Habitability Area, the outside air damper would close, but the HVAC system would not shut down. It would operate in a full recirculation mode.

In areas requiring divisional separation for safe-shutdown, such as in the Reactor Building and the diesel generator equipment areas, active smoke control would be provided. Smoke control would be according to relevant portions of NFPA 92A, to isolate the smoke to one affected division. This active smoke control function requires the system to have the ability to provide outside air and relieve smoke as needed. Accordingly, in these areas, the combination smoke-fire damper would not have a fusible link that could close the damper and compromise the ability to open the damper to relieve smoke in the smoke control mode.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-030

PSAR Sections: 9A

Question Date: March 6, 1998

PSAR Question:

5. **Review of Radioactive Releases while in the Smoke Removal Mode**

   Sources of radioactive material that can be released as a result of a fire will be identified in the formal FHA according to PSAR section 9A.3.1. The acceptance criteria for the fire hazards review in PSAR section 9A.2.4 includes the provision of ventilating systems that use filtration systems to minimize the release of radioactive materials. According to PSAR section 9.4.5, the smoke removal mode for the secondary containment HVAC system bypassses the exhaust filtration system.

   Describe how the formal FHA will evaluate the potential for and consequences of smoke-borne radioactive releases while the ventilation system is in smoke removal mode. Also, discuss the potential for and consequences of radioactive releases due to inadvertently placing the HVAC system in smoke removal mode or inadvertently opening the exhaust filtration system bypass dampers.

PSAR Response:

5. The FHA will evaluate the potential for and consequences of smoke-borne radioactive release by compiling combustible load data for all areas of the RB and CB. High combustible loading in plant areas will be compared to radiation zone drawings to determine areas which present potential for this type of release. The FHA may recommend additional fixed suppression systems if required to minimize the potential for a non-compliant release.

   Regarding the potential for releases due to inadvertent operation of smoke removal mode or inadvertent alignment of bypass dampers, it should be noted that these actions are manual actions. Initiation of smoke removal or exhaust filter bypass will only occur as a conscious action of the plant operator based on alarms from the HVAC system, Fire Protection system, and Radiation Monitoring system. In the smoke removal mode exhaust air radiation monitors will trip the ventilation fans on detection of high radiation. Smoke relief which would bypass the filters in the presence of high radiation alarms would be a manual override of automatic functions. See section 11.5.2 for additional information on the Radiation Monitoring system.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-030

PSAR Sections: 9A

Question Date: March 6, 1998

PSAR Question:

6. **Smoke Migration In Secondary Containment**

   BTP CMEB 9.5-1, Position C.1.b requires the FHA to identify hazards where safety related losses can occur due to lack of adequate access or smoke removal facilities that impede fire extinguishment in safety-related areas. Although the secondary containment ventilation system has a smoke removal mode, this system operates on a divisional basis and not on a fire area basis. As a result, smoke has the potential to migrate to other fire areas containing safety-related equipment of the same division.

   Explain how this potential for smoke migration has been considered regarding the potential to impede fire extinguishment. Will the formal FHA address this issue?

PSAR Response:

6. The HVAC smoke removal mode will be designed to provide pressure differential between divisions as discussed in sections 9.5.1.1 and 9.5.1.1.6. This includes consideration of potentially open divisional boundary fire doors required to facilitate fire fighting. If a fire area is surrounded by fire area(s) of the same division, the FHA will evaluate the potential for impairment of manual fire fighting by considering the combustible load in the area, any potential ignition sources, the expected response time of the detection system, expected access, etc. Based on this evaluation, the FHA may recommend additional fixed suppression systems if required.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-030

PSAR Sections: 9A

Question Date: March 6, 1998

PSAR Question:

7. **Tank and Sump Charcoal Filters**

   PSAR section 9A.5.3 identifies charcoal canister-type filters for process tanks and drain sumps as a special case for the fire hazards review. These filters are not provided with temperature monitoring or fixed suppression systems. No justification for not providing temperature monitoring or fixed suppression systems is provided.

   Do these filters contribute to an exposure fire to safety-related equipment? What fire detection or fire containment systems are used for these filters? Provide justification for the stated position.

PSAR Response:

7. The requested information is part of the longer term design process. At this time, no charcoal canister-type filters for process tanks and drain sumps are present in the Lungmen NI design. If during design activities such filters are identified, the combustible load and potential exposure hazard will be considered in the FHA. Details of the FHA will be provided in the FSAR.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-030

PSAR Sections: 9A

Question Date: March 6, 1998

PSAR Question:

8. **Explosion Hazards**

BTP CMEB 9.5-1, Position C.1.b requires that hazards to safety related equipment related to explosions be identified and appropriate protection provided. PSAR section 9A.2 does not identify any criteria regarding explosion protection.

Identify areas in the plant that pose an explosion hazard. For these areas, describe the criteria and approach to be used to protect safety related equipment from the explosion hazard.

PSAR Response:

8. The requested information is part of the longer term design process. At this time, the only potential explosion hazard present in the Lungmen NI design would be due to potential hydrogen accumulation in primary containment. The Flammability Control System (FCS) is provided to control the potential buildup of hydrogen and oxygen from design basis metal water reaction and radiolysis during a LOCA. See section 6.2.5 for additional information on FCS.

If during detail design activities additional explosion hazards are identified, the combustible load and potential exposure hazard will be considered in the FHA. Details of the FHA will be provided in the FSAR. In the Fire Hazard Analysis due consideration will be give to the areas in the plant that pose and explosion hazard and the measures taken to prevent any impact on safe shutdown equipment.
As a minimum the following areas will be considered.

Oil Field Transformers

Fuel Storage Areas
Fuel Loading Areas
Hydrogen Storage Area
Bulk Gas Storage (either compressed or cryogenic)

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-030

PSAR Sections: 9A

Question Date: March 6, 1998

PSAR Question:

9. **Cross-Divisional Cables on RHR, RCIC, And Containment Isolation Valves**

   PSAR section 9A.5.5 indicates that the electrical circuits for the RHR shutdown cooling outboard isolation valves, the RCIC Main Steam supply outboard isolation valve, and certain outboard containment isolation valves are of a division different that of the line in which the valve is installed. The reason for this is to provide redundancy for isolation capability. As such, these electrical circuits are located in fire areas containing safety-related components of a different division. The justification provided discusses the capability of achieving the valve function if damaged by a fire. However, these circuits are associated circuits of concern. No discussion is provided regarding the protection of upstream power supplies from hot shorts or shorts to ground, nor the potential for propagation the fire to a fire area of another division via this circuit. PSAR section 9A.5.7 presents an analysis of typical circuits of special cases, but excludes the above special cases.

   For the valves identified above, discuss the methods of protecting their electrical circuits.

PSAR Response:

9. The containment isolation valve are powered from redundant divisions to ensure the availability of the containment isolation function during various line break scenarios. The containment isolation function is not required during fires. Dedicated conduit raceways will be used in all cases to ensure availability of these containment isolation valves during the postulated fire scenarios.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-030

PSAR Sections: 9A

Question Date: March 6, 1998

PSAR Question:

10. **Qualifications of FHA Preparer**

    BTP CMEB 9.5-1, Position C.1.a requires the staff for the fire protection program, including the Fire Hazards Analysis preparation and maintenance, to have personnel with training an experience in fire protection and personnel with training and experience in nuclear plant safety.

    Describe the minimum qualifications of the personnel who will prepare the format FHA.

PSAR Response:

10. See section 9.5.1.6.1 regarding Fire Protection Engineer qualifications. Note also that the fire protection and reactor systems engineers performing the NI FHA will meet the requirements of BTP CMEB 9.5.1 Position C.1.b. Also, note that responsibility for the Lungmen NPS

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-031

PSAR Sections: 9.4.1 & 9.1.4.3

Question Date: March 17, 1998

PSAR Question:

There are two Fingers at the end of the Refueling Machine Grapple and an indicator light showing "Engagement" will go on when fingers are closed. But in PSAR there was no indication a "Disengage" indicator light is available. Before Kuosheng's refueling machine was modified, there was only an "Engagement" indicator light. One incident happened when one of the fingers opened and the other one was not for some reason (the indicator light was off) so the operator mistakenly thought the fuel bundle has been released and started operating the Refueling Machine which caused fuel bundle collision as a result. Please clarify if there is "Disengage" indicator light in Lungmen design ? If not, how similar incidents can be avoided ?

Response:

In the purchase specification for the Refueling Machine there is the following requirement for a Fuel Hoist Interlock. The purpose of the interlock is to prevent the type of problem described.

There shall be an interlock to interrupt hoist motion under the following circumstances. Grapple hooks not closed and fuel hoist loaded shall interrupt hoist motion.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-032

PSAR Sections: 9.3.6

Question Date: March 17, 1998

PSAR Question:

1. Section 9.3.6.2 (last paragraph of page 9.3-17) stated that the MSIV signal of the Leak Detection System will close the outboard isolation valve of the instrument air system. Please clarify:

   - Is this indicating the driving air of the outboard MSIV being isolated ?
   - Or it indicating the outboard containment penetration pipe isolation valve in the IAIR system?

2. Section 9.3.6.5 (last paragraph of page 9.3-19) stated that there is a manual controlled motor operated valve located at IAIR containment penetration pipe. Please clarify:

   - How many isolation valves are at IAIR containment penetration pipe ?
   - Will the valves be closed automatically when PCIS signal is present?

PSAR Response:

1. The statement "An MSIV isolation signal from the Leak Detection and Isolation System shall close the IAIR outboard isolation valve." is intended to mean that the IAIR outboard containment penetration isolation valve will close. It is NOT intended to imply that the IAIR supply to the outboard MSIVs will be isolated.

2. There is one IAIR containment penetration. This single containment penetration is equipped with a single outboard motor

operated globe valve and a single inboard soft seated lift check valve.

The outboard motor operated IAIR containment penetration isolation valve will be automatically closed with a signal from the DCIS.(We assume the term "PCIS" in the question should have actually been "DCIS", referring to the Plant Computer or Distributed Control and Information System.)

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: **09-033**

PSAR Sections: **9.5.1**

Question Date: **February 3, 1998**

PSAR Question:

1. Section 9.5.1.1.6 (page 9.5-12), 5th line of the 5th paragraph, the statement "differential pressure ....6.4^{mm} of water" is different from 4.6 ^{mm} of water in SSAR section 9.5.1.1.6. Please explain.

2. Section 9.5.1.2 (page 9.5-17), 1st paragraph, please explain if there is any safety concern when "lubricating oil" is allowed in the CRD pump room. (See SSAR 9.5.1.2, 10th paragraph, which did not mention lub. oil in the CRD pump room)

3. Section 9.5.1.3.1, please explain if the Access control building is part of the service building described in SSAR 9.5.1.3.1 ?

4. Section 9.5.1.3.2 mentioned 130 L/min and 166.7 kpa (ROC requirements) which will not satisfy the 1893 L/min and 448.2 kpa requirements (see SSAR 9.5.1.3.2) of NFPA13. [Note: This is s special consideration when ROC rules and regulations are applied. Please take this seriously]

5. Section 9.5.1.3.5 did not specify the Sprinkler type of the Auxiliary boiler area. Please explain. (see SSAR 9.5.1.3.5, 3rd paragraph...(4))

6. Section 9.5.1.3.7 (page 9.5-22), 4th paragraph, a 25m was mentioned but it is different from the 30.5m in the SSAR 9.5.1.3.7. Please explain.

7. Section 9.5.1.3.9 (page 9.5-23), please supply additional explanation whether the quality class of the Fire Alarm system in the Emergency diesel generator room is class 1E.

Response:

1. Amendment 37, Revision 9 of the Std. ABWR SSAR includes the value

"6.4 mm of water" in section 9.5.1.1.6. This is identical to the value used in Lungmen PSAR section 9.5.1.1.6.

2. The Std. ABWR Fire Hazard Analysis (FHA) is included in the SSAR; Section 9A. Section 9A.4.1.1.15 specifically addresses the CRD pump room, and evaluates the CRD pump lubricating oil hazard. For Lungmen, reference to the CRD pump lube oil was added to Section 9.5.1.2 of the PSAR as an editorial change. This was because the Lungmen FHA is not included in the PSAR (The FHA will be a separate document).

   GE expects the Lungmen CRD equipment to be similar to equipment described in the Std. ABWR SSAR, and does not expect any safety concerns beyond those addressed in the Std. ABWR Fire Hazard Analysis. Actual Lungmen equipment will be evaluated in the Lungmen FHA when design details are available. Results of the FHA will be provided in the FSAR. Also, note that Table 9.5-5 of the Lungmen PSAR indicates automatic sprinkler protection is provided for this area.

3. The Lungmen Access Control Building provides controlled access to each units' radiologically controlled areas and to the respective Control Building. It also provides space for unit specific administrative functions and each units radiochemistry lab. The access Control Building has similar functions to the ABWR Service Building noted in SSAR 9.5.1.3.1 but more limited in scope.

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

4. GE agrees that the wording of PSAR section 9.5.1.3.2 is not clear, and proposes the following revisions to clarify the intent:

   Replace second sentence with the following three sentences:
   "This requirement will meet NFPA 14 wet standpipe flow demand of 1893 L/min at a residual pressure of 448.2 kPa at the most hydraulically remote hose connection in plant buildings. This also meets CMEB 9.5-1, Position 6.c (4) for simultaneous flow of two hydraulically most remote hose connections (approximately 284 L/min each.). In addition, this exceeds the minimum requirements of ROC Fire Protection Code for wet standpipe

minimum flow demand of 130L/min at a residual pressure of 166.7 kPa at the most hydraulically remote hose connection."

Although GE does not expect the above revision to significantly impact Fire Protection System (P16) line sizing, GE will review the P16 line sizing calculations and verify whether any changes are required to the P&IDs and/or the Fire Protection Mechanical Interface Document. Any changes required to these documents will be incorporated in a future revision.

Also, please note that the appropriate NFPA reference for wet standpipe flow and pressure requirements [in accordance with CMEB 9.5.1, Position 6.c (4)] is NFPA 14.

5. The Lungmen Auxiliary Boiler Building is protected with a fixed head automatic suppression system.

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

6. 25m is the maximum distance allowed by the ROCFPC, Article 34 (NFPA 14 allows a maximum distance of 30.5 m). The Lungmen FP design includes additional hose stations in order to satisfy the ROCFPC requirement.

7. The emergency diesel generator (EDG) fire alarm system is not Class 1E, because operation of the fixed fire protection systems is not required to achieve safe shutdown. The local fire panel is, however, provided with primary power from the Vital AC Power System (R13), and is also provided with battery backup in accordance with NFPA 72 and CMEB 9.5.1. The EDG fire alarm system is also designed to be seismically qualified in order to prevent inadvertent operation of the EDG preaction foam-water sprinkler system. This is to ensure that a safe shutdown earthquake (SSE) will not cause failures which could impair the functioning of nearby safety-related structures, systems, and components. See also PSAR section 9.5.1.3.7.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-034

PSAR Sections: 9.3.5

Question Date: April 29, 1998

PSAR Question:

1. As stated in Sec. 9.3.5, there is a RNM not downscale for 3 minutes" requirement to confirm the ATWS condition. However, it is 2 minutes for Kuosheng NPP. Please explain the base for the 3 minutes.

2. It is mentioned that the design of SLC should take otential leakage" into consideration. Please describe the potential leakage considered in Lungmen SLC design.

3. It is mentioned that 25% margin of boron concentration is added to allow potential leakage and imperfect mixing. Please explain why 25% margin is enough to coves leakage and mixing issues.

Response:

1. GE analyzed the limiting transients identified in NEDO-24222, Assessment of BWR Mitigation of ATWS (supporting document used for ATWS rule) for Kousheng and the ABWR. The NEDO assessment (with alternate 3 design with automatic SLC initiation) included a two (2) minute time delay to allow for ARI and other actions to cause rod insertion or to allow for operator intervention in the event of spurious initiation, thus avoiding unnecessary boron injection.

   For the Lungmen ABWR, the ATWS mitigation design features include RPT, RFCS RIP Runback logic, ARI, FMCRD Run-in, FW run-back, and auto SLC injection. SLC is designed to auto initiate after a three (3) minute time delay if the SRNM was not downscale. SLC will not auto initiate if the SRNM is downscale, which indicates that either the ARI or FMCRD run-in was successful and the reactor is shutdown. For the FMCRD run-in (slow scram) with a failure of ARI, the

FMCRDs take approximately 145 seconds to be fully inserted electrically. Additionally there is some delay time between the FMCRD initiation signal and start of FMCRD motion which including full-in stroke time forms the basis of the 3 minute time delay. The 3 minute time delay accommodates the delay and insertion time of the FMCRD, ARI actuation, other actions to cause rod insertion or allows for operator intervention in the event of spurious SLC auto initiation to avoid unnecessary boron injection.

2. Potential leakage refers to leakage of borated reactor coolant outside the reactor vessel, for example into sumps, such as identified or unidentified reactor coolant system leakage. By design the SLC system shall be able to bring the reactor from full power to a subcritical condition without control rod movement, at any time with the reactor in the most reactive xenon-free state and maintain it.

3. The 25% margin of boron concentration, added to allow for potential leakage and imperfect mixing, is a design requirement on all BWRs and the currently operating ABWRs. The minimum boron concentration of 850 ppm by weight of boron compensates for various reactivity effects in the reactor, with additional margin conservatively required to account for uncertainties such as possible leakage of the coolant, imperfect mixing of the boron solution in the core, and dilution of the borated reactor coolant by unborated RHR Shutdown Cooling water.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-035

PSAR Sections: Ch 9.5.1

Question Date: April 20, 1998

PSAR Question:

Please describe approaches which will be taken to avoid the occurrence given on NRC Information Notice 97-01: improper electrical grounding results in simultaneous fires in the control room and the safe-shutdown equipment room.

PSAR Response:

NRC Information Notice IN 97-01 describes the simultaneous occurrence of two fires which resulted from a single ground fault within an isolation transformer utilized in an uninterruptible power supply (UPS) essential lighting arrangement. The affected system was designed with ground connection on the neutral leg of the inverter instead of grounding the neutral leg of the power source (regulating, isolation transformer) in accordance with industry practice.

Based on a review of the NRC information notice stated above and the inherent design features of the UPS power distribution of the Lungmen Nuclear Unit 1 & 2, It is determined that a similar situation does not exist. This determination is based on the fact that the 120 VAC UPS system is designed for ungrounded (floating) operation with ground detection devices installed at each distribution panel. This includes the regulating transformers, inverters, distribution panels, and equipment powered by the Vital AC power system. Therefore, while a single ground fault does not prevent the system from operating continuously, it is instantly detected and alarmed to initiate corrective action.

Therefore, adherence to the following established design will ensure that a similar situation is not inadvertently created for Lungmen

Nuclear Units 1 & 2:

- Ensure consistency of the ungrounded status of the CVCF power distribution system. If equipment design necessitates an internally grounded component, the vendor will be required to provide isolation transformer with appropriate winding connections to ensure that a ground fault current path is not created because of the subject component.
- Provide overload protection on all transformer secondary winding leads.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-036

PSAR Section: 9.2.4

Question Data: APRIL 21,1998

PSAR Question:

1. For section 9.2.4.2.3 Sewage treatment system, please explained the treat capacity and the standard requirements of sewage water discharge.
2. In section 9.2.4.6 only mention about the hydrostatically test of drainage discharge piping. Please specify that whether PSW system supplying water pressure test follow the local regulator standards.

PSAR Response:

1. The treatment capacity of the wastewater treatment plant is estimated to be 630m^3/day. The standard requirement of sewage water discharge shall be in compliance with the Taiwan/ROC EPA's 1998 discharge criteria.

2. The PWS system will be hydrostatically tested to the applicable ROC Plumbing code requirement or to the following requirement whichever is more conservative:
   Test the subject piping to static pressure of 50 psig (345 kpa) above operating source and allow to stand for 4 hours. Leaks and loss in test pressure constitute defects that must be repaired.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-037

PSAR Sections: Ch. 9

Question Date: May 27, 1998

PSAR Question:

1. When will the auxiliary fuel pool be built ? If it is built at the same time as the plant, then it will be idle for more than 15 years after completion. Please explain how the equipment maintenance will be carried out during this time period.

2. The auxiliary fuel pool is next to the diesel generator. Is the location proper ? Its operation safety should be evaluated and consideration should be given to the related nuclear safeguard works of the auxiliary fuel pool during the diesel generator O&M.

3. Is it the best choice to have dry transportation ? It seems spent fuel transportation will be simpler and safer if water duct is adopted (such as the Kuosheng design).

4. Sections 9.3.9, 9.3.10 and 9.3.11 are blank. Are they reserved for certain purposes or omissions ? Please clarify.

5. There was no explanation of the relevent codes and standards that are used for the design of the auxiliary fuel pool. Please explain.

6. (9.1.2.1.3) only stated the design of RACK but no explanation was given for the concrete and rebar structures of auxiliary fuel pool. Please supply this information.

7. (9.1.2.1.5) did not show the material specifications of auxiliary fuel pool. Please supply this information.

8. (9.1.3.2.1) The water temperature of auxiliary fuel pool is controlled to be 49 oC (120 oF) which is higher than the design condition of 90 oF ~ 110 oF in Section 6.3.2.2 of ANSI/ANS 57.7. Please explain.

9. (9.1.3.2.2) Figures 9.1-15, 9.1-16 and Table 9.1-13 are missing. Please supply.

10. (9.1.3.2.2) Is it sufficient to have two 0.6MW heat exchangers ? Is it sufficient that the pump is designed for 225 m3/h ? Please clarify or provide calculation sheets for evaluation.

11. Is it proper to classify piping and components as Quality Group C ? Please explain.

12. (9.1.4.2.10.3) Please explain whether Dewatering and filling with Inert Gas are necessary for the Transfer Cast after fuel loading. Also, sealing operation should be explained too.

Response:

1. The Auxiliary Fuel Pool Building will be constructed during the project construction stage. For equipment to be procured and installed during the plant construction stage, the operating and maintenance procedures (including preventive and corrective requirements) will be developed and available prior to shipment and installation of the equipment. Maintenance of this equipment will be done periodically, after installation based on manuals and procedures, and this will keep equipment in good working conditions.

2. The location of the diesel generator in the next to the Auxiliary Fuel Pool causes no concerns to the O&M of the diesel generator based on the following considerations:

   Fuel transfer from the Reactor Building to the Auxiliary Fuel Pool will only occur during normal plant operation so O&M on the diesel generator can be performed during planned outages with no concern with fuel transfer.

   The radiation levels in and around the Auxiliary Fuel Building are expected to be low since the fuel will have decayed for approximately 15 years prior to transfer. The radiation levels around the Auxiliary Fuel Building should be unrestricted radiation levels less than 0.0025mSv/h. Primary source of radiation around the Auxiliary Fuel Building is expected to be gamma shine from the turbine buildings.

The access to the diesel generators is totally separate from the Auxiliary Fuel Pool and associated equipment. Since the transport takes place inside the TPC controlled Lungmen plant boundary, there is no nuclear safeguard concern.

3. The use of Spent fuel casks to move the fuel from the Reactor Building Spent Storage Pool to the Auxiliary Fuel Building Spent Storage Pool is considered to be the best choice because the spent fuel can be transferred during normal plant operation and have no impact on refueling outage activities. The use of a fuel transfer tube for this application is not practical due to the distances between the buildings. In the Kuosheng plant the inclined fuel transfer tube passes between adjacent buildings. The spent fuel transport distances from the Reactor Building to the Auxiliary Fuel building are 200 m for Unit 1 and 260 m for Unit 2.

4. The PSAR Section numbering system follows the US ABWR SSAR which includes Section 9.3.9 for the Hydrogen Water Chemistry (HWC) System, Section 9.3.10 for the Oxygen Injection (O2I) System and Section 9.3.11 for the Zinc Injection (ZIS) System. Since HWC, O2I and ZIS are not incorporated in the Lungmen design sections 9.3.9, 9.3.10 and 9.3.11 have been omitted from the Lungmen PSAR.

5,6. and 7. As noted in Section 3.8.4.2.6 of the Lungmen PSAR all Auxiliary Fuel Building structures are designed to the same codes and standards as the Reactor Building and the details are provided in Section 3.8.4.2.1. Additional information is provided in Section 3.8.4.1.6.

8. ANSI/ANS 57.7 was written to provide guidance for Independent Spent Fuel Storage Installations (ISFSI) of the water cooled pool type.

In other words a standalone spent fuel pool. This type of facility would be licensed separately under 10 CFR 72. Since the Auxiliary Fuel Pool (AFP) for Lungmen is not fully independent and is being licensed as part of the 10 CFR 50 license for the plants, ANSI/ANS 57.7 is not entirely applicable.

On the other hand Standard Review Plan (SRP) 9.1.3 document is the design basis for Spent Fuel Pool Cooling and Cleanup System (SFPC). Per this document the maximum SFPC pool temperature at maximum heat load shall be below 60 °C (140 °F). Hence by controlling the AFPC water temperature to 49 °C (120 °F) the SRP requirement is met.

9. As noted on Page 9.1-77 for Figure 9.1-15, the figure is proprietary and is provided in a separate binder, GE PROPRIETARY INFORMATION - VOLUME 2. As noted on Pages 9.1-78 and 9.1-79 of Figure 9.1-16, the figure will be provided in FSAR. Regarding Table 9.1-13, a page for the table will be added showing, "to be provided in FSAR," and the List of Tables will list Table 9.1-13 on Page 9.0-xi.

10. Yes it is sufficient to have two 0.6 MW heat exchangers and pumps designed for 225 m^3/h. They will provide sufficient cooling capability to keep the auxiliary fuel pool temperature below 49 °C (120 °F). The final analysis results will be included in FSAR.

11. Standard Review Plan (SRP) 9.1.3 document is the design basis for Spent Fuel Pool Cooling and Cleanup System (SFPC). Based on SRP and since Auxiliary Fuel Pool (AFP) and Auxiliary Fuel Building (AFB) ventilation and filtration system is seismic Category I and Quality Group C, it is proper to classify piping and components as Quality Group C.

12. For fuel shipping on site, the cask should be dewatered but whether it would need to be inerted or seal welded will depend on the cask transfer system used. For example, a typical IF-300 cask system uses a railcar or a heavy haul vehicle for transport of the cask covered by a retractable enclosure and a cooling system. A typical transfer process starting in the reactor building using a heavy haul vehicle is as following:

(1) Roll the vehicle into position.
(2) Perform a radiation survey.
(3) Remove cask hold down pins and install lifting equipment.
(4) Attach the handling yoke to the crane, then attach the yoke to the cask.
(5) Lift and rotate the cask to a vertical position, then raise and place the cask in the decontamination area.
(6) Clean and fill the cask and remove all but 4 cask closure head sleeve nuts.
(7) Remove the cask from the cleaning area and place the cask in the fuel transfer cask pit.
(8) Remove cask closure head nuts and head.
(9) Load fuel into the cask with the refueling machine.
(10) Inspect seal surfaces and replace cask closure head.
(11) Raise the cask, replace closure nuts, and rinse off cask.
(12) Place cask in decontamination area and decontaminate the cask.
(13) Secure cask closure head.
(14) Typically at this time for irradiated fuel shipments, the cask is flushed, drained, dried and the cask is filled with helium.
(15) The cask is then taken from the cleaning area and placed on the transport vehicle and secured.
(16) The cask is moved to the Auxiliary Fuel building where the process is repeated with on exception. There is no specific decontamination area. The cask will be placed directly into one of the pits for preparation of the unloading process.

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-038

PSAR Sections: Ch. 9

Question Date: May 27, 1998

PSAR Question:

Lungmen is a twin unit design. Please explain that besides auxiliary fuel building, what other facilities are shared by both units. Has consideration been given to the design of structures, systems and components that are important to safety for these shared facilities so that the safety functions will not be adversely effected by the failure of any one unit to safely shutdown the other unit?

Response:

The Lungmen design conforms to the USNRC General Design Criteria (GDC)(See PSAR Section 3.1). In regard to the question, the design conforms to Criterion 5 (GDC-5), which requires that consideration be given to sharing of the structures, systems and components that are important to safety so that, in the event of an accident in one unit, the safety functions will not be significantly impaired in the other unit. This is discussed fully in PSAR section 3.1.2.1.5.2. Also, please refer to the responses to Track No. 19-027 (Part 3) and its follow-on question in Round 2 and to Track No. 09-030 (Parts 1, 2 and 3).

Both the non-safety related Circulating Water Pumphouse and the safety related Service Water Pumphouse are shared between both Lungmen units.

The safety related service water system is designed such that each of the safety related service water pumps is located in its own flood protected enclosure, physically and electrically isolated from the other pumps. There are three trains of Service Water provided for each unit. Each service water train includes two full capacity pumps, physically and electrically independent of each other, which take suction on their dedicated service water pit. Each service water suction well includes a dedicated screen, accessed from a common gallery. The redundant pumps of each train feed a common header (one of three per unit) which is installed in a dedicated pipe tunnel independent of the other service water trains. The pipe tunnels are arranged into three tunnels for each unit. These design arrangements ensure that, in the event of an accident in one unit, the safety functions of components for the unaffected unit will not be significantly impaired.

The non-safety related circulating water systems and turbine building service water systems for both units utilize a common pump house. This non-nuclear safety, seismic category II, quality group D structure houses the circulating water intake screens, the six circulating water pumps for each unit, and the turbine building service water pumps for each unit. The circulating water and turbine building service water pumps for both units are located in unit-specific sections of a shared open gallery in this pumphouse. The traveling water screens for these pumps are accessed from this common gallery. Failure of this non-safety structure or non-safety systems will not significantly impair the safety functions of components for the unaffected unit.

Section 3.1.2.1.5.2 will be revised as indicated in the response

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-039

PSAR Sections: 9.1.4.2

Question Date: May 28, 1998

PSAR Question:

Section 9.1.4.2.10.2.1.1 stated that the new fuel bundles will be stored in receiving warehouse first but there is no description of the receiving warehouse or whether the warehouse structure, fire protection and safeguard control will meet the requirements of Article 29 of the Atomic Energy Implementation Rules. Please clarify.

Response:

There is no receiving warehouse available for the Lungmen Project. Instead, the incoming new fuel will be delivered directly to the refueling floor near the new fuel storage vault. PSAR subsection 9.1.4.2.10.2.1.1 will be revised as follows:

"The incoming new fuel will be delivered directly to the refueling floor near the new fuel storage vault where the new fuel will be examined for damage during shipment.

The new fuel arrives in its wooden crate at the Reactor Building via the rail and truck entry door. The wooden crate dimensions are approximately 813 x 813 x 5486 mm. Each crate contains one metal RA container which contains two fuel assemblies. The shipping weight of each unit is approximately 13.35 kN. The inner RA container is removed from the reusable wooden crate and is then lifted to the refueling floor by the RB crane. The covers are removed from the inner RA container and the RA container is raised to vertical. The new fuel bundles are removed and moved to the new fuel inspection stands with the RB crane. The new fuel bundles are visually inspected and checked dimensionally. The fuel bundle is channeled with a new channel. The new fuel bundles are then placed in the fuel pool or the new fuel storage vault using the RB crane."

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-040

PSAR Sections: 9.1.4.3 and 9.1.5.5

Question Date: May 28, 1998

PSAR Question:

1. In Section 9.1.4.3 of SSAR, it was stated that "To satisfy NUREG 0554, the equipment handling components over the spent fuel pool are designed to meet the single-failure-proof criteria" but in Lungmen PSAR this paragraph was modified to include only the 9.81KN hoist handling component that will meet the single-failure-proof criteria.

2. Comparing SSAR Section 9.1.5.5 and Lungmen PSAR, similar wordings have been modified so that only 9.81KN hoist and RIP hoist meet the single-failure-proof criteria.

3. Based on the above differences, please explain what kind of hoists which can pass over the top of the spent fuel pool ? and which hoists can not meet NUREG 0554 single-failure-proof criteria requirements ? If there is any, what is the replacement design or method ?

Response:

NUREG-0554 deals with the handling of critical loads. For reactor servicing equipment the critical load for the refueling machine has been defined as equivalent to the handling of a fuel assembly. If the hoist can handle a fuel assembly or a component heavier than a fuel assembly, it is considered to handling a critical load.

Refueling Machine

The Refueling Machine has three (3) hoists. They are:

1. The 9.81 kN hoist used to handle the RIP impeller shaft, RIP Diffuser/Stretch Tube, and the Combination Fuel Support/Control Rod

Grapple. NUREG-0554 and NUREG-0612 apply to the 9.81 kN hoist because this hoist handles loads which exceed the weight of a fuel assembly (definition of a heavy load/critical load defined above).

2. The main fuel hoist which is used to handle fuel. NUREG-0554 applies to the fuel hoist. NUREG-0612 does not apply to the fuel hoist.

3. The 4.71 kN hoist (which is limited to 227 kg) is used to handle small vessel internal components. The requirements of NUREG-0554 and NUREG-0612 do not apply as the light components carried by this hoist are not critical loads nor heavy loads. However we have requested the following supplemental requirements apply to this small hoist. These are:

   a. A design margin of 15% shall be applied to items in the lift path that are determined to be subject to degradation due to wear.

   b. When a redundant load path does not exist, individual parts of the vertical hoisting system components, which include the hoisting cable interface, and cable receiving systems shall be designed to support a static load of 200% of the Rated Load Capacity. This is in addition to the design margins required by CMAA No. 70.

   c. When a redundant load path exists, the standard design margins required by Reference CMAA No. 70 apply.

   d. The hoist has two full capacity brakes so arranged as to ensure that either one will prevent an uncontrolled descent of the load. An electrically operated, fail-safe, automatic brake will be provided which is sized to bring the hoist and load to a safe, vibration-less stop from full speed. The brake will be capable of holding 150 percent of normal hoisting load when power to motor is cutoff, or in case of complete power failure

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-041

PSAR Sections: 9.5.1

Question Date: May 28, 1998

PSAR Question:

- 9.5.1.3.5
  The sprinkler systems in the Reactor Building and the wet stand pipe systems in the Reactor and Control Building are designed in compliance with ASME B31.1.
- 9A.2.3(7)
  The Nuclear Island design utilizes an ANSI B31.1 standpipe system in all Seismic Category I buildings.

The codes used for the wet standpipe system in the above two paragraphs are different. Which one is correct ? Please explain.

Response:

9A.2.3 (7) contains an incorrectly worded reference. The standpipe system design discussed in 9A.2.3(7) is designed to ASME B31.1.

Although ASME B31.1 is a standard that has been adopted by ANSI (American National Standards Institute), it is not appropriate to refer to this standard as "ANSI B31.1". Therefore, the reference to "ANSI B31.1" in section 9A.2.3(7) will be changed to "ASME B31.1".

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 09-042

PSAR Sections: 9.5.7 and 9.5.6

Question Date: June 26, 1998

PSAR Question:

9.5.7 It was stated in the section on Cooler for D/G lubricant that the cooling water was provided by Jacket Cooling Water but from P&ID 9.5.6, the cooling water for the Cooler was from the coolant discharge of the Jacket water heat exchanger (it should be the RBCW system, according to section 9.5.5). Obviously, there is difference between the P&ID and the description in this section. Please clarify.

9.5.6 It was stated that the function of the D/G Starting Air Dryer is simply to control the Dew Point to within supplier's suggested value but no mention was given to whether it will comply with SRP 9.5.6 section II.4.J requirements (i.e., lower than 50 °F under controlled 70 °F environment or, 10 °F lower than the lowest expected surrounding temperature).

Response:

9.5.7

The lube oil cooler is cooled by RBCW, as shown on the P&ID. The following changes to the PSAR will be made to clarify this.

The referenced sentence in PSAR Section 9.5.7 will be changed to read,

"Cooling water for the coolers comes from the jacket cooling water *system* (Subsection 9.5.5)."

The RBCW distribution piping associated with the diesel is a part of this system.

The second paragraph of PSAR Section 9.5.5.2 will be changed to read, "Each diesel generator unit is supplied with a complete closed loop cooling system mounted integrally with the engine generator package. Included in each cooling package are a jacket water heater and keep-warm pump, temperature-regulating valve, lube oil cooler, motor and/or engine-driven cooling water pumps, jacketed manifold and a jacket water cooler. The jacket water cooler and lube oil cooler are furnished with cooling water from the essential portion of the RBCW system, or in the case of the Swing Emergency Diesel Generator, a cooling loop which contains pumps and an air-to-water heat exchanger. RBCW supply is from the same division as that of the diesel generator served."

The third paragraph of PSAR Section 9.5.5.2 will be changed to read,

"The jacket cooling water passes through a three-way temperature control valve which modulates the flow of water through or around the jacket water heat exchangers (coolers), as necessary, to maintain required water temperature. Jacket water cools the turbocharger, the governor, the air cooler, and the exhaust manifold. The three-way valve, whose service is crucial, is designed and qualified as stated in Subsection 9.5.5.1."

9.5.6 The referenced sentence in PSAR Section 9.5.6 will be changed to read,

"The dryer will be capable of controlling the dew point to the more limiting requirement of either:

(1) Not more than 10° C (50° F) when installed in normally controlled 21.1° C (70° F) environment, otherwise the starting air dew point should be controlled to at least 5.56° C (10° F) less than the lowest expected ambient temperature.

(2) Dew point recommended by the diesel generator manufacturer."