編 輯 說 明

一、本附件係原能會執行台電公司龍門計畫初期安全分析報告書審查時，委請美國 MPR 工程顧問公司就耐震設計、儀控系統、火災危害分析及嚴重核子事故之防範與事故後之減輕等四部份併行審查後，所提出之審查報告，列爲原能會審查結論報告附件之一。

二、前項委請 MPR 公司併行審查之部份，其涵蓋章節如下：

1.第三章 3.7 節及 3.8 節(耐震設計)

2.第七章(儀控系統)

3.第九章 9A 節(火災危害分析)

4.第十九章(嚴重核子事故分析)

三、本附件內容概分爲（一）審查結論、（二）問題和答覆、（三）附錄等三部份。原報告以英文撰寫，經原能會專案小組審閱後，摘其重要結論如後表，謹供參閱。

四、併行審查過程中所提之各項問題與台電公司之答覆說明，均已納入原能會所編「審查問題與答覆彙編」資料中。

五、本附件中所提之審查結論或建議事項，悉經原能會專案小組審閱後，併入審查結論主體報告中。

美國 MPR 工程顧問公司併行審查結論摘要表

項　目	內　容　摘　述
耐震設計	一、本項審查目的在確認核四廠的耐震設計是否符合相關法規、規範與標準，耐震設計中所使用分析方法與程序是否正確。審查內容包含耐震設計所用之阻尼值、設計反應譜等參數選用的正確性、反應器廠房與圍阻體、控制廠房與其它地震一級結構的分析方法、負載的訂定與組合的周詳性，結構耐震的選用準則的適切性等。 二、審查結果認為，上述項目在初期安全分析報告書所提內容，可以接受。惟詳細計算的模式與結果，必須在未來終期安全分析報告書審查階段詳細說明，並建議屆時要特別注意下列各項： 1.地震一級結構耐震分析中，所使用結構阻尼值與結構應力的關聯性，要詳細說明，並正確反映所使用阻尼值之正確性。 2.核四廠廠房結構具有對稱性，可使用二維模式以簡化分析方法，但要詳細說明二維與三維模式在頻率、參與係數與模態三方面之比較，以確定二維模式之適用性。 3.耐震次系統的動態時間歷程積分法中，安全釋壓閥沖放負載之時間歷程可以不考慮±15%的誤差而直接引用試驗值，但要有詳細的說明。 4.反應器廠房、控制廠房、輔助燃料廠房、安全廠用海水泵室耐震分析中結構要考慮滑動、翻轉與浮動等效應之安全因素。在土壤與結構的互制分析中要考慮防水層間的摩擦效應。
儀控系統	一、本項審查目的在確認核四廠安全相關系統之儀控設計及功能符合多重性、獨立性、可測試、單一故障準則及失效安全準則等，是否符合法規要求；並具有足夠的安全餘裕，能確保在任何狀況下均能提供運轉人員足夠資訊以安全操控機組。 二、審查期間曾要求提供核四廠安全儀控軟體發展計畫，並提出主蒸汽管路高輻射信號設計及遙控停機盤上氮氣控制裝置不對稱等主要問題。 【註：有關上述問題之處理，扼要說明如下： 1.對於軟體發展計畫問題，台電公司已補送軟體構型管理、軟體驗証與確認作業書等相關文件，以供審查。

儀控系統	2.至於移除主蒸汽管高輻射跳機信號之設計問題，根據以往沸水式機組之營運經驗顯示，此項跳機功能並無實質必要，故於進步型機組中將之取消。此項設計修改並經美國核管會審核通過(詳奇異公司安全分析報告 NEDO-31400A)。 3.另對遙控停機系統設備不對稱性問題，台電公司將於第二遙控停機盤面上增加氮氣控制裝置，以爲解決方案。】 上述問題經澄清處理後，第七章之內容及設計已符合美國聯邦核能技術法規(10CFR)、安全分析報告審查標準(SRP)及儀控工業標準(如 IEEE)等法規的要求。 三、在建廠期間，建議應注意下列事項： 1.應加強儀控系統裝機及測試的品質保證查証工作，尤其對數位儀控軟體的發展，更應落實驗證與確認的工作，以確保儀控系統的高品質。 2.對於遙控停機盤儀控設備的採用，應注意其與主控室之間互具有多樣性的關係，以避免共因失效。
火災危害分析	一、核能電廠火災防護系統之設計需符合美國聯邦核能法規 10CFR 50 附錄 A 之一般設計準則第 3 條和第 5 條、10CFR 50.48(防火法規)以及 10CFR 50 附錄 R（核能電廠消防作業計畫）之要求。 二、本項之審查係依據上述法規執行，審查結果證實核四廠火災危害分析所使用之準則及方法，可以接受，而相關內容之說明，如煙控系統、火災抑制系統以及火災時輻射外釋之控制等，符合法規之要求。 三、未來在終期安全分析報告書審查階段，除了對具體內容需再評估外，另建議對以下二項可能性繼續追蹤防範： 1.反應器廠房二次圍阻體外發生火災時，應考量煙塵侵入二次圍阻體通風系統時，是否會影響安全區域之人員與設備。 2.消防系統動作後，滅火劑侵入其他防火區之影響評估。

嚴重核子事故分析	一、本項審查重點係針對核四廠在設計上如何預防嚴重核子事故之發生，以及一旦在事故發生後，如何緩和事故所造成的影響，進行相關內容審查。 二、審查所依循之標準，主要爲美國核管會針對嚴重核子事故所發行之法規文件，包括：SECY-90-016、SECY-93-087、及 NUREG-1503 等。 三、經審查核四廠因應嚴重核子事故之設計，以及相關之定量模擬分析結果後，證實核四廠在預期暫態未急停(ATWS)、電廠全黑事故(SBO)等之預防方面以及一旦發生嚴重核子事故時，對氫氣燃燒、爐心熔渣與混凝土作用等現象之緩和方面，在設計上均已作充分考量而可接受。惟在未來終期安全分析報告書審查階段，建議應注意下列各項： 1.建議發展一套核四廠嚴重核子事故之審查標準。 【註：美國核管會針對嚴重核子事故所發行之 SECY-90-016、SECY-93-087、及 NUREG-1503 等法規文件，已涵蓋嚴重核子事故之審查項目及接受標準，當可做爲審查本章之依據，故實質上已有可資引用之審查標準。】 2.建議參考美國核管會的作法，對於電廠全黑事故、高壓熔融物噴出(HPME)等嚴重核子事故現象，利用不同之計算程式(如:MELCOR)，在核四建廠期間，再執行獨立之驗證分析。 3.建議在本章中補充評估地震及颱風之影響。 【註：核四廠初期安全分析報告書附錄 A 之章節 AC 及 AF 中，已分別針對這兩種肇因進行相關安全分析，原能會在審查期間亦要求台電公司針對地震所造成嚴重核子事故及圍阻體損壞的機率進行分析，其結果顯示發生之機率是在合理可接受的範圍內。】 4.對於圍阻體之耐壓等定量分析方面，台電公司應提出更具體完整之分析。

MPR
ASSOCIATES INC.
ENGINEERS

Safety Evaluation Report (SER) for Partial Review of Lungmen Nuclear Power Station Preliminary Safety Analysis Report

MPR-1946
Revision 0

Submitted by

MPR Associates, Inc.
320 King Street
Alexandria, VA 22314

June 1998

Prepared for

Atomic Energy Council, Republic of China (ROC-AEC)
Taipei, Taiwan

Table of Contents

PART I

1.0 Introduction 1-1

2.0 Overall Conclusions of the Partial Review of Lungmen PSAR 2-1

3.0 SER on Lungmen PSAR Sections 3.7 and 3.8, "Seismic Design and Structures" 3-1

4.0 SER on Lungmen PSAR Chapter 7, "Instrumentation and Control" 4-1

5.0 SER on Lungmen PSAR Appendix 9A, "Fire Hazards Analysis" 5-1

6.0 SER on Lungmen PSAR Chapter 9, "Severe Accident Analysis" 6-1

PART II

Appendix A Formal Questions and Answers on Sections 3.7 and 3.8 ..... A-1

Appendix B Formal Questions and Answers on Chapter 7 ..... B-1

Appendix C Formal Questions and Answers on Appendix 9A C-1

Appendix D Formal Questions and Answers on Chapter 19 ..... D-1

PART III

Addendum 1 Summary of Open Items E-1

Addendum 2 Recommendations for Future Actions F-1

PART I

1

Introduction

MPR Associates, Inc. (MPR) has completed a third-party independent review of the Preliminary Safety Analysis Report (PSAR) of the Taiwan Power Company (TPC) Lungmen Nuclear Power Station Project. The effort was performed for the Republic of China Atomic Energy Council (ROC-AEC).

The objective of the review was to assist the ROC-AEC in assuring (1) the plant has been designed in conformance with relevant codes and standards and (2) the plant has been designed by using valid methodology and criteria and by conducting analyses and evaluations in sufficient depth.

To achieve the objective of the proposed PSAR review, MPR assembled a team of senior staff within our company as well as senior industry experts outside of our firm with extensive expertise and experience in all technical areas within the proposed scope of work. The review team consisted of MPR Associates, EQE International, PLG (a division of EQE) and Winston & Strawn. MPR Associates provided overall project management and technical review of Lungmen PSAR Chapter 7, "Instrumentation and Control" and Appendix 9A, "Fire Hazards Analysis." EQE performed the technical review of Sections 3.7 and 3.8, "Seismic Design and Structures." PLG performed the technical review of Chapter 19, "Severe Accident Analysis." Winston & Strawn performed overall licensing and regulatory review.

This report presents the results of our review in the form of a Safety Evaluation Report (SER). The format and wording were selected as if they were coming from the ROC-AEC. For example, where the SER refers to "the Staff," it means "the ROC-AEC Staff." The following work was performed for each of the sections and chapters:

The relevant TPC PSAR submittals were reviewed. The primary reference for this review was the USNRC Standard Review Plan (SRP). All other applicable codes and standards, regulations, guidance, branch technical positions, and specifications were used and are referenced in the individual SER sections.

- Questions were compiled during the process of the review for TPC to respond. These questions were discussed with and agreed upon with ROC-AEC prior to being sent to TPC.
- A preliminary Safety Evaluation Report (SER) was submitted to ROC-AEC for review.

- A question and answer meeting was conducted at ROC-AEC headquarters to discuss the review results/conclusions and the issues remaining to be clarified.
- A final SER, was prepared with appendices addressing the raised questions, TPC's answer to each question, and our comments/suggestions on TPC's answers.

Section 2 presents the overall conclusions of the review. The four SER sections are presented in Sections 3, 4, 5 and 6 of this report. Appendices A, B, C and D present the questions proposed by MPR to TPC (through ROC-AEC) on their PSAR, and the answers received from them. Additional questions were asked directly by ROC-AEC and are not included specifically in this report.

2

Overall Conclusions of Partial Review of the Lungmen PSAR

All sections and subsections of Lungmen PSAR Sections 3.7 and 3.8, Chapter 7, Appendix 9A and Chapter 19 were thoroughly reviewed. Over 100 questions were generated from this review, with answers received from TPC on most all of them. While there are still a handful of open items associated with our review, no items represent significant issues which should prevent the construction permit from being issued for Lungmen.

The overall conclusion of this partial review is that the commitments made by TPC, represented by the PSAR sections and responses to our questions, should not prevent ROC-AEC from issuing the construction permit (CP) for Lungmen Nuclear Power Station.

3

SER on Lungmen PSAR Sections 3.7 and 3.8, "Seismic Design and Structures"

3.7 SEISMIC DESIGN

The staff reviewed the seismic design adequacy of the Lungmen NPS presented in the PSAR using SRP Sections 3.7.1 through 3.7.4 as the basis and considered TPC's responses to questions posed by the staff. All sub-sections of the PSAR were reviewed.

The PSAR primarily presents seismic design methodology. Only limited seismic design details were available for review. Further review of the Lungmen NPS seismic design adequacy will be required when such details are available.

3.7.1 Seismic Design Parameters

The seismic design ground response spectra for the Lungmen NPS are normalized mean plus one standard deviation design spectra for rock sites . The SSE spectra are anchored to peak horizontal and vertical ground accelerations of 0.40g. These ground response spectra were previously approved by the staff. Further review of these spectra based on PSAR Section 3.7.1 was not performed.

Three SSE acceleration time-histories, one for each of the two horizontal and one vertical directions, were generated. Response spectra for these time-histories were computed at frequency intervals suggested by Table 3.7.1-1 of SRP Section 3.7.1, plus control points of the design spectra, at 0.5%, 1%, 2%, 5%, 7%, and 10% damping. Comparisons of the computed response spectra to the SSE design ground response spectra satisfy enveloping criteria specified in SRP Section 3.7.1. Statistical independence of the three acceleration time-histories was demonstrated by calculation of correlation coefficients less than the maximum acceptable value of 0.16 recommended by Reference 6 of Regulatory Guide (R.G.) 1.92.

Comparisons of power spectral density (PSD) functions for the acceleration time-histories with targets that are compatible with the design ground response spectra were not provided in the PSAR submitted for review. In response to the staff's question, TPC described the method used to generate target PSD functions compatible with the site-

specific design ground response spectra. This method is in accordance with SRP requirements and is acceptable to the staff. TPC also provided comparisons of the PSD functions for the ground acceleration time-histories with these target PSD functions for review in response to the staff's request. Acceptability of the target PSD functions and those for the ground acceleration time-histories was confirmed by the staff's review. TPC will include the comparisons of PSD functions in PSAR Figures 3.7-24, 3.7-25, and 3.7-26.

The damping ratios used in the analysis of the Lungmen NPS seismic Category I structures and components comply with the SSE damping ratios specified in R.G. 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," and R.G. 1.84, "Design and Fabrication Code Case Acceptability, ASME Section III Division I." The PSAR does not identify how structure stress levels and damping values are to be correlated as specified by the SRP. The use of maximum damping values specified by R.G. 1.61 is acceptable for seismic analysis to obtain seismic loads for structure design since structure stresses cannot exceed permissible limits without these damping values being achieved. However, compatibility between structure stress levels and structure damping values for seismic response analysis to obtain in-structure response spectra for equipment design must be demonstrated. A procedure to provide this compatibility is presented in Section 3.1.2.2 of ASCE 4-86, "Seismic Analysis of Safety-Related Nuclear Structures." This item remains open.

Damping values for the foundation rock . obtained by in-situ dynamic testing are presented in PSAR Table 2.5-17. This table also lists damping values for the crushed rock backfill. These values are strain-dependent in accordance with SRP Section 3.7.1 requirements. In response to the staff's question, PSAR Section 3.7.1.3 will be amended to reference this table as the basis for rock damping values.

Descriptions of the supporting media for the seismic Category I structures, including foundation embedment depth, depth of soil over bedrock, soil layering characteristics, design groundwater elevation, structural foundation dimensions, total structural height, and soil properties, are not included in PSAR Section 3.7.1.4. In response to the staff's question, PSAR Section 3.7.1.4 is to be amended to reference the data presented in PSAR Sections 2.5.4.2.3 and 2.5.4.2.4. Strain-dependence of the foundation rock and

crushed rock backfill shear modulus in PSAR Table 2.5-16 will also be referenced in PSAR Section 3.7.1.

The staff concludes that (1) the synthetic ground motion time-histories, (2) the target PSD functions compatible with the site-specific ground response spectra and PSD functions for the synthetic ground motion time-histories, (3) and structure and soil damping values for seismic analysis to obtain structure seismic design loads satisfy applicable criteria. Descriptions of the supporting media for the seismic Category I structures are provided as required by the SRP. However, compatibility between structure stress levels and structure damping values for seismic response analysis to obtain in-structure response spectra for equipment design has not been demonstrated. This item remains open and is subject to future verification.

3.7.2 Seismic System Analysis

The staff's review of the seismic analysis of the seismic Category I SSCs includes the seismic analysis methods and acceptance criteria used for the Lungmen NPS seismic Category I structures and the reactor pressure vessel (RPV) design. The radwaste building, which is a seismic Category IIB structure, is included in this review because it is to be designed to maintain structural integrity for OBE seismic loads in accordance with R.G. 1.143.

The Lungmen PSAR does not describe seismic analysis of the turbine building. TPC provided additional information concerning the main steam lines and turbine building in response to the staff's questions. The main steam lines are to be routed from the reactor building through the main steam tunnel above the control building and into the turbine building. The MS-BOP main steam line piping, bypass line, and condenser are required to remain functional during and after an OBE and a SSE. The MS-BOP main steam lines, turbine bypass valve lines including the turbine bypass valve steam chests and all branch lines 65 mm pipe size and larger, up to and including the first valve that is either normally closed or capable of automatic closure during all modes of operation, are to be dynamically analyzed to demonstrate their structural integrity under OBE and SSE loading conditions. The turbine building is a seismic Category IIA building. Such structures are designed according to ROC Building Code "Medium Seismicity Zone"

criteria and are analytically checked to determine that they will not collapse when subjected to an SSE and extreme environmental loads. The two horizontal and one vertical SSE ground motions are considered to act simultaneously when checking the structural integrity of the turbine building.

Dynamic analysis of the turbine building is necessary to determine SSE seismic loads for verification that structural integrity is retained and to obtain seismic input for dynamic analysis of the main steam lines and other associated components. Acceptability of the turbine building seismic analysis cannot be determined since the PSAR does not describe this analysis. Also, the staff cannot determine if the SSE structural integrity check of the turbine building is acceptable since detail on criteria for this check, such as analysis methods, load combinations, and acceptance criteria, have not been provided. These items remain open.

Seismic analysis of the Lungmen NPS seismic Category I structures and RPV are performed by the response spectrum or time-history methods. Time-history analyses are performed in either the time-domain or frequency-domain. Equipment supported at different elevations in a building are analyzed by either (1) response spectrum analysis using the worst single floor response spectrum from the set of floor response spectra at the various floors applied at all floors or (2) multiple support excitation analysis methods. Descriptions of the methodologies for the seismic analysis methods are provided in the PSAR. These methodologies are acceptable to the staff.

The criteria to verify that an adequate number of masses or dynamic degrees of freedom are included in the structure models were not identified in the PSAR. In response to the staff's question, TPC noted that the number of masses is considered to be adequate when (1) it is equal to at least twice the number of significant modes in the vibration direction of interest or (2) when additional degrees of freedom do not result in more than a 10 percent increase in responses. These criteria are consistent with SRP Section 3.7.2 requirements and are acceptable.

Methods for calculating natural frequencies and response loads of the seismic Category I buildings are identified in the PSAR. The calculated frequencies and seismic loads are not reported.

Methods for analytical modeling of the seismic Category I structures, RPV and internals, and radwaste building are presented in the PSAR. In general, three-dimensional models are used for seismic analysis of the seismic Category I structures. However, two-dimensional models may be used if coupling between the two horizontal directions is negligible. Three-dimensional models are considered by the staff to be preferable. In response to the staff's questions on the use of two-dimensional models, TPC committed to establishing the equivalence between the two- and three-dimensional models by comparing only their calculated frequencies and modal participation factors. The staff considers it appropriate to compare mode shapes as well. This item remains open.

Criteria for decoupling a subsystem from the supporting system are presented in the PSAR. These criteria satisfy SRP Section 3.7.2 requirements. In lieu of these criteria, the PSAR cites an alternative approach to account for the effect of system-subsystem interaction. This alternative approach is presented in PSAR Reference 3.7-8. The alternative method was found to be acceptable by the staff based on review of the cited reference.

Seismic response of the RPV and reactor internals is obtained by a coupled dynamic analysis with the reactor building. Development of the RPV and internals model is summarized. These analytical modeling procedures are acceptable to the staff.

The seismic analysis of the radwaste building is performed using a dynamic model with lumped masses having six degrees of freedom at the floor levels. The masses are connected by beam elements having equivalent stiffness properties. This methodology for development of the radwaste building analytical model is acceptable to the staff.

Soil-structure interaction (SSI) analysis is performed using Computer Program SASSI. Further information on the SSI analysis is not provided in the PSAR. Recent investigations have indicated that the use of the same nodes to model the excavated soil and the interior foundation of the belowgrade structure, while implicitly permitted by the SASSI user's manual, can lead to erroneous results in certain instances. In response to the staff's question, TPC noted that SASSI models for the Lungmen NPS seismic Category I structures do not use the same nodes for the excavated soil and interior foundation models, thus avoiding this potential error. TPC's response is acceptable to the staff.

PSAR Section 3.7.2.4 does not describe the treatment of structure-soil-structure interaction. TPC responded that the Lungmen seismic Category I structures are to be founded on competent rock and structure-soil-structure interaction effects are expected to be insignificant. This response is acceptable to the staff.

Floor response spectra are obtained from time-history analyses of the buildings. In-structure acceleration time-histories are used to calculate response spectra in the two horizontal and one vertical directions for each of three earthquake ground motion components input separately. These spectra are combined by the square-root-of-the-sum-of-the-squares (SRSS) method to obtain the total co-directional floor response spectra. Alternatively, co-directional floor response spectra are obtained from time-history analysis with simultaneous input of the three earthquake ground motion components when the ground motion time-histories are statistically independent. These procedures for calculation of floor response spectra, as well as frequency intervals at which the spectral accelerations are calculated, are in accordance with SRP Section 3.7.2.

Seismic responses due to the three earthquake ground motion components are combined by the SRSS method of R.G. 1.92 or the 100-40-40 method of ASCE 4-86. Alternatively, when time-history analyses are performed using three statistically independent earthquake ground motion components, the total response may be obtained by algebraically adding the co-directional responses calculated separately for each component at each time step. In addition, the combined response from time-history analysis with the three statistically independent earthquake ground motion components input simultaneously may be used directly.

Use of the SRSS method and the time-history analysis procedures satisfy SRP Section 3.7.2 requirements. The 100-40-40 method, while not explicitly cited in R.G. 1.92 or SRP Section 3.7.2, was accepted by the USNRC in the final safety evaluation report for a certified standard advanced PWR design in NUREG-1462. These methods for combination of seismic responses due to the three earthquake ground motion components are acceptable to the staff.

Modal responses calculated by response spectrum analysis are combined in accordance with R.G. 1.92. Responses of high frequency modes are considered in accordance with

Appendix A of SRP Section 3.7.2. These modal combination procedures satisfy SRP Section 3.7.2 requirements and are acceptable.

Criteria to prevent seismic Category IIA, IIB, and IIC structures from affecting the integrity/function of seismic Category I SSC are specified in PSAR Section 3.7.2.8. These criteria are acceptable except, as noted above, further detail on criteria for the seismic Category IIA turbine building is needed.

The floor response spectra are broadened to account for uncertainties in structure and soil material properties and uncertainties in the analytical models. The spectral peaks associated with the structure frequencies are broadened by ±15% if parametric variation studies are not performed. If parametric variation studies are performed, structural frequencies are broadened by a minimum of ±10%. The peak shifting method of ASME Section III, Appendix N may also be used. These procedures are in accordance with SRP Section 3.7.2, R.G. 1.122, and R.G. 1.84.

Procedures for inclusion of torsional effects in conjunction with two-dimensional analytical models are described. An accidental eccentricity equal to 5 percent of maximum building dimension at each floor was used to calculate seismic shear for distribution to the lateral load-resisting structural elements. These procedures are in accordance with SRP Section 3.7.2.

In PSAR Section 3.7.2.14, a structure is defined as stable against seismic overturning when the ratio of the potential energy needed to overturn the structure to the maximum kinetic energy of the structure during the SSE is no less than 1.1. This approach complies with the guidelines of SRP Section 3.7.2 and is acceptable.

Composite modal damping ratios for a structure consisting of materials with different damping ratios were obtained using the strain energy of individual components as a weighting function. This procedure is consistent with the guidelines of SRP Section 3.7.2 and, therefore, is acceptable.

The methodology and criteria for seismic analysis of the seismic Category I SSCs are acceptable to the staff, with the following exceptions:

1. Dynamic analysis of the turbine building is necessary to determine SSE seismic loads for verification that structural integrity is retained and to obtain seismic input for dynamic analysis of the main steam lines and other associated components. Acceptability of the turbine building seismic analysis cannot be determined since the PSAR does not describe this analysis.

2. Criteria for the SSE structural integrity check of the turbine building have not been described in sufficient detail to determine if these criteria are acceptable.

3. In addition to frequencies and modal participation factors, mode shapes obtained by two- and three-dimensional models should also be compared to verify their equivalence.

3.7.3 Seismic Subsystem Analysis

The seismic analysis procedures and methods applied to the Lungmen seismic Category I subsystems that were not explicitly included in the seismic analyses of the seismic Category I structures were reviewed. Such subsystems include all seismic Category I piping and supports, cable trays and supports, conduit and supports, above-ground tanks, buried piping and tunnels and structural elements that support other seismic Category I items, the dynamic effects of which could affect the seismic response of the supported items (e.g. steel platforms, frame type support structures, etc.). Seismic qualification of seismic Category I mechanical and electrical equipment is presented separately in PSAR Sections 3.9 and 3.10.

3.7.3.1 Seismic Analysis Methods

The PSAR states that the analysis methods applied include:

1. Dynamic Analysis

2. Testing

3. Equivalent static load methods of analysis

4. A combination of 1 and 2

5. A combination of 2 and 3

Dynamic analysis was conducted using the either the response spectrum or time history approach. Time history analysis is performed by either the mode superposition method or the direct integration method. Input motions to subsystems are defined as response spectra or time histories obtained from the results of the structural response analyses. Time history input at support points is developed in accordance with the techniques described in PSAR Section 3.7.2.1. In-structure response spectra are developed as described in PSAR Sections 3.7.2.5 and 3.7.2.9.

When the time history analysis method is used the input time history is broadened by plus and minus 15% of the integration time step. Force-time histories for non-seismic dynamic loads such as safety relief valve blowdown are not broadened on the basis of conservatism observed between dynamic tests and analysis. When response spectrum analysis is used, the broadened and smoothed in-structure response spectra are used.

Input motion to testing is defined by the broadened and smoothed in-structure response spectra at the location of the equipment. Testing of components is addressed in PSAR Sections 3.9.2.2 and 3.10.

The staff previously requested the justification for not broadening the force-time history for non-seismic loads such as SRV blowdown. TPC responded that tests have been performed which confirm the conservatism of the analytical results. Therefore, for the SRV blowdown loads, the calculated force-time histories are not broadened.

The staff agrees that, if test data confirms the conservatism of the calculated force-time histories, that adding further conservatism by broadening is not warranted. However, the response to the question is general and specific test reports and numerical ratios of test vs. analysis results have not been identified. The PSAR should identify the specific test data source and specify the degree of conservatism. This item remains open.

The staff requested a description and verification of the computer codes used in the evaluation of subsystems. TPC identified the codes to be used for subsystem analysis and further clarified that descriptions and verifications for these codes are presented in PSAR Sections 3D and 3D.1, respectively. TPC's response is considered to resolve the staff's question.

The staff finds that the methods of seismic analysis are in compliance within the requirements of SRP and are acceptable for the design of Lungmen subsystems, with the exception noted above.

3.7.3.2 Determination of Number of Earthquake Cycles

The PSAR states that one SSE and five OBEs of 10 cycles each, as recommended in SRP Section 3.9.2, are used in the seismic evaluation of subsystems. Alternatively, fractional peak cycles equivalent to maximum peak cycles for five OBEs may be used in accordance with Appendix D of IEEE-344. Regulatory Guide 1.100 accepts IEEE-344 and the staff finds that the PSAR definition of the number of seismic fatigue cycles is in accordance with regulatory guidance.

3.7.3.3 Procedures Used for Modeling

The PSAR provides a description of procedures used for modeling piping systems (PSAR Section 3.7.3.3.1), modeling of equipment (PSAR Section 3.7.3.3.2), field location of supports and restraints (PSAR Section 3.7.3.3.3), and analysis of frame type supports (PSAR Section 3.7.3.3.4). The modeling of piping covers the selection of mass points, selection of spectrum curves, analysis of decoupled branch piping lines, selection of input time histories, modeling of pipe supports, modeling of special engineered pipe supports and response spectra amplification at support attachment points.

The staff requested a technical justification for the criteria for selection of relative anchor motion for decoupling of branch pipe. TPC responded that the decoupled branch pipe must have a moment of inertia less than 1/25 of the run pipe which makes it very flexible. Further, the decoupled branch pipe must not have a support near the intersection with the run pipe.

It is noted by the staff that in PSAR Section 3.7.3.8.9, "Effect of Differential Building Movements, "the spacing of the branch pipe support from the run pipe is clarified as being no less than 1/2 of the recommended dead weight support span specified in ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, Table NF 3611.1." This minimum support distance and the restriction on 0.16 cm relative displacement of branch and run pipe supports is acceptable. However, the description of design and analysis of branch piping currently provided in PSAR Sections 3.7.3.3.1.4 and 3.7.3.8.9 should be consolidated so that the complete criteria are clearly defined in one subsection.

In PSAR Section 3.7.3.3.4, "Analysis of Frame Type Supports," coefficients of friction are given for calculating the friction forces on the support. The staff requested justification for the values stated. TPC has responded stating that the technical justification will be provided in the FSAR and that the corresponding sentence in Section 3.7.3.3.4 of the PSAR would be replaced to state that "[t]he static coefficient of friction that will be used in the analysis will be provided in the FSAR with a technical justification." This change to the PSAR is acceptable to the staff.

The staff agrees with the rest of the modeling descriptions contained in PSAR Section 3.7.3.3.

3.7.3.4 Basis of Selection of Frequencies

The PSAR states that the frequency range between 0.25 and 33 Hz covers the range of the broad band response spectrum used in design. Subsystems with frequencies greater than 33 Hz are treated as seismically rigid and analyzed accordingly. Frequencies less than 0.25 Hz are not considered since systems this flexible are not encountered in the plant. These criteria for qualification of seismic Category I subsystems are acceptable.

3.7.3.5 Use of Equivalent Static Load Methods of Analysis

Refer to PSAR Section 3.7.3.8.5

3.7.3.6 Three Components of Earthquake Motion

The methods described in the PSAR are in compliance with R.G. 1.92 and are acceptable.

3.7.3.7 Combination of Modal Response

Section 3.7.3.7.1 of the PSAR states that when the response spectrum method of analysis is used, modal responses below the cutoff frequency of 33 Hz will be combined by SRSS and that closely spaced modes will be combined by either the grouping method, the ten percent method or the double sum method. The staff finds this to be consistent with Regulatory Guide 1.92 and acceptable.

PSAR Subsection 3.7.3.7.2 presents methodologies to account for high frequency modes and assure that sufficient mass is participating in the dynamic solution such that the addition of any higher modal response will not increase the overall response by more than 10%. The staff finds the stated practice to be consistent with the SRP and to be acceptable.

3.7.3.8 Analytical Procedure

The procedures described in this section of the PSAR defines rigid subsystems with rigid supports (PSAR Section 3.7.3.8.2), rigid subsystems with flexible supports (PSAR Section 3.7.3.8.3), and flexible subsystems. Guidance is provided for static analysis (PSAR Section 3.7.3.8.5) and dynamic analysis (PSAR Section 3.7.3.8.6) of the above subsystem categories. The staff finds that the static and dynamic analysis practices stated are consistent with the SRP and are acceptable.

PSAR Section 3.7.3.8.7 provides guidance on damping for piping (R.G. 1.84 Code Case N-411), other subsystems (R.G. 1.61), strain-weighted damping as specified in PSAR Section 3.7.2.15 and Alpha-Beta damping for direct integration. The staff finds the damping criteria specified to be consistent with the SRP and to be acceptable.

PSAR Section 3.7.3.8.8 describes the methodology to evaluate and combine the responses to differential building movements with those of inertia. It is stated that

stresses in the subsystem produced by differential building movements are secondary. It is further stated that the peak dynamic response to inertia loading, which is considered to produce primary stress, and the peak response due to differential building motion are not expected to occur at the same time. Therefore the primary and secondary loads are combined by the SRSS method.

Two questions were asked by the staff. The first question pointed out that Section II.9 of the SRP requires that responses of multiply-supported equipment and components due to inertia effects and relative displacements be combined by absolute sum, rather than by SRSS, and requested a technical justification for the deviation. The second question pointed out that the combination of primary and secondary responses was based on philosophy for piping systems but that for piping supports and other subsystem support design, loads that produce secondary stresses in piping are treated as though they produce primary stress in supports. Since part of the argument for SRSS combination of inertia and building displacement loads was based upon the primary/secondary stress argument, it was requested that a description be provided as to how the forces/stresses in piping supports due to seismic inertia and support point displacement are combined for support design and the justification for the combination.

TPC responded to the first question stating that the dynamic and pseudo-static (displacement) contributions are time-inconsistent. Their peak values do not occur at the same instant in time and a more representative combined peak response is provided by SRSS combination of peak dynamic and peak pseudo-static values. The staff agrees that SRSS combination of responses to inertia and support point displacements is more reasonable than absolute sum combination and is consistent with the regulatory guidelines for SRSS combination of modal responses. The staff agrees that SRSS combination of inertia and support point responses, where required by the load combination criteria, is acceptable.

TPC responded to the second question stating that pipe support loads due to seismic inertia and support point displacements are combined by SRSS. PSAR Section 3.9.3.4, Section 3.9.3.4.1, Table 3.9-1, and Table 3.9-2 state that (1) loading combinations for pipe supports for various operating conditions correspond to those used for design of the supported pipe and (2) seismic anchor motions are to be combined in accordance with

NUREG-0484, Revision 1. This NUREG permits the combination of dynamic loads by SRSS. The staff considers TPC's response to be acceptable.

PSAR Section 3.7.3.8.9 describes criteria for design of small branch and small bore piping. The branch piping criteria in PSAR Section 3.7.3.3.1.4 is complementary to the description of evaluation of branch piping in PSAR Section 3.7.3.8.9. The combination of both sections are technically acceptable. However, as stated in the staff comments on PSAR Section 3.7.3.3.1.4, the description of design and analysis of branch piping should be consolidated so that the complete criteria are clearly defined in one subsection.

The subsection further provides discussion on the use of the small bore piping handbook and restrictions on its use. It is stated that the small bore piping handbook is accepted by the regulatory authorities for piping at other nuclear plants. Use of proven and regulatory approved small bore piping handbook methods is acceptable to the staff as long as the handbook is consistent with the ASME code rules in effect at the time of the piping design.

In PSAR Section 3.7.3.8.10, "Multi-Supported Equipment and Components with Distinct Inputs," guidance is provided on the use of envelope response spectra and independent support motion (ISM) response spectra. If ISM spectra are used, the responses of excitation in each direction for each support or group of supports are combined by SRSS. ISM spectra methods are not used in combination with ASME Code Case N-411 damping. Alternatively, independent support motion time history input may be used in a time history analysis. The guidance provided for evaluating subsystems with distinct support motion inputs is acceptable.

3.7.3.9 Multi-Supported Equipment and Components with Distinct Inputs

This title is redundant with PSAR Section 3.7.3.8.10.

3.7.3.10 Use of Constant Vertical Static Factors

Guidance is provided on the use of static analysis in lieu of dynamic analysis for calculating dynamic response to vertical seismic input. The staff finds that the guidance provided is consistent with regulatory guidelines in the SRP.

3.7.3.11 Torsional Effects of Eccentric Masses

Reference is given to the treatment of eccentric masses in piping analysis in PSAR Section 3.7.3.3.1.2. The staff finds this to be acceptable guidance.

3.7.3.12 Buried Seismic Category I Piping and Tunnels

It is stated that details of methods used for determining the design parameters associated with seismically induced transient relative deformations are to be included in the FSAR. Industry references for guidance purposes are provided as References 3.7-2 and 3.7-7 of the PSAR. The cited references for guidance are acceptable to the staff.

3.7.3.13 Interaction of Other Piping with Seismic Category I Piping

Guidance is provided for analysis of seismic Category I piping connected to seismic Category II piping. If an anchor does not isolate the two categories of piping, the coupled system model must extend beyond two seismic restraints in each of the three orthogonal directions. Additionally, if it is not feasible or practical to isolate the two systems, the seismic Category II piping must be designed to withstand the SSE. The staff finds this guidance to conform to industry practice and to be acceptable.

3.7.3.14 Seismic Analysis of Reactor Internals

Seismic analysis of reactor internals is discussed in PSAR Section 3.7.2.3.2. The description of the modeling, including the fluid coupling effect, in PSAR Section 3.7.2.3.2 appears to be the standard modeling methodology and procedure use by GE in most of their BWR reactor internal design and is acceptable to the staff.

3.7.3.15 Analysis Procedure for Damping

Reference is made to procedures in PSAR Sections 3.7.2.15 and 3.7.3.8.7. The staff requested information on the types of analyses that would be performed to account for differences in damping for OBE and SSE. TPC responded that the OBE and SSE dynamic analyses are performed using the values contained in PSAR Table 3.7-1 which are consistent with Regulatory Guide 1.61. Damping is described in PSAR Section

3.7.8.7. The staff finds that the damping guidelines conform to Regulatory Guide 1.61 and ASME Code Case N-411 which is accepted with limitations by Regulatory Guide 1.84 and that the values stated are acceptable.

3.7.3.16 Methods for Seismic Analysis of Above-Ground Tanks

General methods of analysis of the tank response considering the impulsive mode, including soil-structure interaction, and the convective mode are described. Guidance is given on evaluating the uplift due to overturning moment and the potential for buckling. Guidance is general except for the specific damping value of 0.5% to be used for the convective mode.

The staff requested the damping value that would be used for the impulsive mode of the tank contents. TPC stated that the PSAR would be modified to state that "damping values used to determine the spectral acceleration for the impulsive mode of the fluid-tank system will be based upon the system damping associated with the tank shell material as well as with the soil-structure interaction." The values will be provided in the FSAR. This response is acceptable to the staff.

3.7.4 Seismic Instrumentation

The seismic instrumentation system specified for the Lungmen NPS in PSAR Section 3.7.4 is acceptable and meets GDC 2 and 10 CFR Part 100, Appendix A. TPC meets these requirements by requiring installation of instrumentation that is capable of adequately measuring the effects of an earthquake.

TPC meets the requirements of 10 CFR Part 100, Appendix A, and GDC 2 by providing instrumentation that is capable of measuring the effects of an earthquake. The installation of the specified seismic instrumentation as specified in PSAR Section 3.7.4.1 in the reactor containment structure, other Lungmen NPS Category I structures, and the free field constitutes an acceptable program for recording data on seismic ground motion as well as data on the amplitude, frequency, and phase relationship of the seismic response of major structures and systems in the event of an earthquake. A readout of pertinent data from the various seismic instruments will yield sufficient information to

guide the operator on a timely basis to determine if the level of earthquake motion ground requiring plant shutdown has been exceeded. Data obtained from such installed seismic instrumentation will be sufficient to ascertain that the seismic analysis assumptions and the analytical models used in the seismic design of the Lungmen NPS were adequate and that allowable stresses have not been exceeded under conditions when continuity of operation is intended. The staff finds the design for seismic instrumentation acceptable.

In view of the time and effort required to determine if the level of earthquake ground motion requiring plant shutdown has been exceeded, plant operating procedures that define specifically what constitutes a significant exceedance of the level of earthquake ground motion requiring plant shutdown should be established. TPC has included his information in Section 3.7.4.4 of the PSAR. This is acceptable.

The SRP has not been formally revised to reflect the new OBE definition in 10 CFR 50, Appendix S. However, holders of operating licenses or construction permits issued prior to the implementation date of the active regulatory guides may voluntarily implement the methods described. The Lungmen NPS seismic instrumentation meets the guidance of the following regulatory guides: (1) R.G. 1.12, Revision 2, March 1997, (2) R.G. 1.166, March 1997, and (3) R.G. 1.167, March 1997. PSAR Section 3.7.4 consequently meets the intent of 10 CFR 50, Appendix S.

3.8 Design of Seismic Category I Structures

The staff reviewed the TPC design commitments to Lungmen Units 1 and 2 seismic Category I structures which are presented in Section 3.8 of the PSAR. The primary bases of the staff's review are:

- NUREG-0800, Standard Review Plans (SRP) Sections 3.8.1 through 3.8.5
- Federal Regulations, 10 CFR 50
- Relevant USNRC Regulatory Guides (e.g., R.G. 1.29, R.G. 1.70, R.G. 1.136, R.G. 1.142, R.G. 1.143)
- Applicable codes and standards (e.g., ASME Section III, Divisions 1 and 2, ACI-349, ANSI/AISC N-690)
- TPC's responses to the staff PSAR review questions

The PSAR Section 3.8 presents primarily TPC's commitments to the design of Lungmen seismic Category I structures. The PSAR contains no discussions on the design analyses results or calculations.

Like the Standard Review Plans, the format of Section 3.8 of the PSAR in general was keyed to the Standard Formats and Contents of Safety Analysis Reports for Nuclear Power Plants, R.G. 1.70. The staff used the Standard Review Plans as a guidance for the PSAR review. TPC's commitments to the design of seismic Category I structures in order to meet the requirements in the applicable regulations, codes, and standards were reviewed by the staff.

3.8.1 Concrete Containment

The containment is designed as a low leakage reinforced concrete cylindrical shell structure with an internal steel liner made of carbon steel as a leak tight membrane, except for wetted surfaces where stainless steel or carbon steel with stainless steel cladding will be used. The containment is divided by the diaphragm floor and the reactor pedestal into an upper drywell chamber, a lower drywell chamber, and a suppression chamber. The top slab of the containment is an integral part of the fuel pool with the pool girders rigidly connected to the containment top slab and the Reactor Building (RB) wall. The containment is enclosed by and structurally integral with the Reactor Building through the RB floor slabs. The containment wall, top slab, RB floor slabs, and foundation mat are constructed of cast-in-place reinforced concrete. The right circular cylinder containment wall is 2.0 m (6 ft 7 in.) thick with an inside radius of 14.5 m (47 ft 7 in.) and a height of 29.5 m (96 ft 9 in.) measured from the top of the foundation mat to the bottom of the containment top slab. The containment foundation mat is 5.5 m (18 ft) thick and is continuous with the RB foundation mat.

PSAR Section 3.8.1 provides descriptions of the containment (Section 3.8.1.1), applicable codes, standards and specifications (Section 3.8.1.2), loads and load combinations (Section 3.8.1.3), design and analysis procedures (Section 3.8.1.4), structural acceptance criteria (Section 3.8.1.5), material and quality control procedures (Section 3.8.1.6), and testing and in-service surveillance requirements (Section 3.8.1.7) for the containment structure. Significant review findings, resolutions of the findings based on TPC's response, and open items resulted from the PSAR review are presented below.

The containment is designed to resist various combinations of dead loads, live loads, environmental loads, including those resulting from wind, typhoon, and earthquakes, normal operating loads, and loads generated by a postulated Loss-of-Coolant Accident (LOCA) such as CO, CH, VLC, PS and AP. The development of the safety relief valve (SRV) loads and hydrodynamic loads resulted from a LOCA will be provided in the FSAR (PSAR Section 3.8.1.3.1). The design loads, load combinations, and load factors considered for the containment structure as listed in PSAR Table 3.8-1 are in compliance with the guidelines of SRP Section 3.8.1. The staff noted that the load

combination No. 8 in PSAR Table 3.8-1 used a load factor of 1.0 for the SRV loads which is not consistent with the 1.25 factor required in the appendix to SRP Section 3.8.1. TPC responded by stating that knowledge gained from plants tests of the SRV loading has confirmed that the inherent margins in the SRV load definition are sufficiently conservative, thus a more realistic factor of 1.0 as specified in the 1989 Edition of ASME Code Section III, Division 2, Subsection CC is technically adequate. This is acceptable.

The containment structure is analyzed by the use of the linear elastic finite element computer program NASTRAN. The containment structure and the RB layout utilizes an integrated structural system. By taking advantage of symmetry of the containment and the RB about the centerline of the plant parallel to the fuel pool girders and the steam tunnel, only 180° of the structure was modeled using assemblage of beam elements and plate elements (PSAR Section 3.8.1.4.1.1). Details of the Containment and Reactor Building finite element models are not provided in the PSAR and will be supplied with the FSAR. Static analyses will be performed to calculate structural element forces with the finite element model subjected to both axisymmetric and nonaxisymmetric design loads. The staff raised questions on the treatment of nonaxisymmetric loads when they are applied to the 180° finite element model to simulate the state of stress and strain in the actual containment structure. TPC responded by stating that for nonaxisymmetric loads the boundary displacements for nodes at the plane of symmetry follow the anti-symmetry conditions. This approach is acceptable.

The dynamic loads on the suppression pool boundaries resulted from the actuation of SRVs will be calculated based upon the methodology which will be presented with the FSAR. In response to staff's question on the definition of appropriate dynamic load factors for hydrodynamic loads, TPC will revise the second last sentence in Page 3.8-9 of the PSAR Section 3.8.1.4.1.1.2 to "The attenuated pressures and equivalent static loads will be calculated based on the methodology to be presented in the FSAR."

The PSAR Section 3.8.1.4.1.1.3 states that the state of stresses and behavior of the containment wall around major penetrations are determined by the use of analytical numerical techniques considering both effects of concrete cracking and thermal strains. In response to staff's review questions concerning the use of the 180° global finite

element model to calculate liner stresses and the stresses in the containment wall around major penetrations, and the effects of concrete cracking, TPC responded by stating that (1) all major penetrations are effectively considered in the 180° model, (2) the liner stresses will be obtained directly from the global finite element model with finer meshes at penetrations, and (3) effects of concrete cracking are accounted for in the calculations by program SSDP-2D which calculates the stresses of concrete and rebars assuming that concrete cracks under tension. Section forces and moments used in the SSDP-2D analysis are obtained from the global finite element model. This analysis approach is found to be acceptable.

The design, fabrication, construction, and testing of the containment are in accordance with Subsection CC of the 1989 Edition of ASME Code, Section III, Division 2. This edition of ASME Code is referenced in 10 CFR 50.55a, and is acceptable. In the review of Safety Analysis Report of General Electric (GE) ABWR (GESSAR), the USNRC staff raised a concern that the code specified allowable tangential shear stress of 0.2fc' is too high for the design of reinforced concrete containments. Subsequently, GE revised the allowable tangential shear. This revised allowable tangential shear stress was accepted by the USNRC. The staff found the allowable tangential shear stress in the Lungmen PSAR Table 3.8-2 properly reflects this revision. However, units of fy and fc' are not clearly specified in the table. In response to the staff question, TPC will change the PSAR by adding a footnote to Table 3.8-2 to specify that fy and fc' must in MPa units. Thus, the upper bounds of tangential shear strength provided by the containment shell orthogonal reinforcing steel shown in Table 3.8-2 are acceptable.

In PSAR Section 3.8.1.7.1, it is stated that TPC will perform the structural integrity test (SIT) of the containment in accordance with the provisions of Article CC-6000 of ASME Code, Section III, Division 2 (1989 Edition) and Regulatory Guide 1.136, after completion of the containment construction. This is acceptable. TPC is required to provide details of the test and the instrumentation as required by the ASME Code, Section III, Division 2, Subsection 6212 for such a test to the staff for review and approval.

The criteria used in the analysis, design, and those proposed for construction of the concrete containment structure to account for anticipated loadings and postulated conditions that may be imposed upon the structure during its service lifetime are in

conformance with established criteria, codes, standards, guides, and specifications and are acceptable. Items that will be provided in the FSAR and should be reviewed by the staff when available are methodology of development of hydrodynamic loads due to SRV and LOCA and the calculations of attenuated pressures and equivalent static loads.

3.8.2 Steel Components of the Reinforced Concrete Containment

The major steel components of the concrete containment, as stated in PSAR Section 3.8.2, consist of personnel air locks, equipment hatches, penetrations, and the drywell head. These components are designed for the same loads and load combinations as those used in the design of the concrete containment shell to which these components are attached. These components will be fabricated, and tested as Class MC components in accordance with the 1989 Edition of Subsection NE of ASME Code, Section III, Division 1. In PSAR Section 3.8.2.4.1.4, TPC stated that the drywell head consisting of shell, flanged closure, and drywell head anchor system will be analyzed. The stresses including discontinuity stresses induced by the combination of external pressure or internal pressure, dead load, live load, thermal effects and seismic loads, will be evaluated in accordance with Subarticles NE-3130 and NE-3200 of ASME Code Section III, Division 1 (1989 Edition). The staff found that the design approach and the use of design code are consistent with the guidelines of SRP Section 3.8.1. This is acceptable. No design calculations were reviewed in the PSAR review.

TPC further stated in Section 3.8.2.4.1.4 of the PSAR that the compressive stress within the knuckle region caused by the internal pressure and the compression in other regions caused by other loads are limited to the allowable buckling stress values in accordance with Subarticle NE-3222 of ASME Code Section III, Division 1 (1989 Edition) or Code Case N-284. This approach is consistent with the USNRC's position on shell buckling as stated in Appendix E of NUREG-1503. This is acceptable.

The criteria used in the analysis, design, and those proposed for construction of steel components of the concrete containment to account for anticipated loadings and postulated conditions that may be imposed upon the structures during its service lifetime are in conformance with established criteria, codes, standards, guides, and specifications and acceptable to the regulatory staff.

3.8.3 Concrete and Steel Internal Structures of Concrete Containment

In the Lungmen design, the internal structures inside the containment include the reinforced concrete diaphragm floor, the reactor pedestal, the reactor shield wall, the drywell and equipment pipe support structure (DEPSS), and other structural components. The diaphragm floor separates the upper drywell from the suppression pool. The reactor pedestal is a composite steel and concrete structure. The pedestal consists of two concentric steel shells tied together by vertical steel diaphragms. The regions formed by the steel shells and the vertical diaphragms are filled with concrete. The reactor shield wall is supported by the reactor pedestal and surrounds the reactor pressure vessel. The reactor shield wall is of similar construction as the reactor pedestal. The DEPSS is a steel structure providing supports for piping, pipe whip restraint, mechanical equipment, electrical equipment, and general access platforms and stairs.

PSAR Section 3.8.3 provides the applicable codes, standards and specifications, loads and load combinations, design and analysis procedures, structural acceptance criteria, material and quality control procedures, and testing and in-service surveillance requirements for the internal structures.

The containment internal structures are designed to resist those loads described in PSAR Section 3.8.1.3 for the containment structure. These loads and their effects are considered in the design of internal structures as applicable. The design loads, load combinations, load factors, and the associated acceptance criteria for the concrete and steel internal structures are listed in Tables 3.8-5 and 3.8-6 of the Lungmen PSAR. They are in compliance with the guidelines of SRP Section 3.8.3 and applicable codes and are, therefore, acceptable.

The major codes used in the design of the concrete internal structure and the steel internal structure are American Concrete Institute (ACI) Standard 349 (1990 edition) and ANSI/AISC N-690 (1984 edition). The USNRC staff expressed concerns about the design requirements for the steel embedments in ACI-349. As a result, exceptions to the use of ACI 349, Appendix B on steel embedments are presented in the Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design (NUREG-1503). These exceptions have been incorporated in the Lungmen

PSAR as Table 3.8-10, which is properly referenced in the PSAR. Limitations to ANSI/AISC N-690 for design of nuclear facility safety-related structures are included in the Lungmen PSAR as Table 3.8-9. Thus, the use of ACI 349 and ANSI/AISC N-690 standards is acceptable.

The containment concrete and steel internal structures are designed to resist various combinations of dead and live loads, accident-induced loads (including pressure and jet loads), and seismic loads. The load combinations used cover those cases likely to occur and include all loads that may act simultaneously. The containment internal structures are designed and proportioned to remain within the limits in accordance with SRP Section 3.8.3 for the various load combinations. These limits are based on ASME Code, Section III, Division 2, ACI Standard 349, and ANSI/AISC Standard N-690 for concrete and steel internal structures, respectively, modified as appropriate for load combinations that are considered extreme.

For the analysis and design, the diaphragm floor, the reactor pedestal, and the reactor shield wall were included in the combined finite element model of the containment structure and the Reactor Building. The computer code NASTRAN was used for analyses of the finite element model. The analysis and design procedures are essentially the same as those used for the containment analysis and design. As discussed in Section 3.8.1 of this report, they are acceptable.

The DEPSS is a three-dimensional space frame steel structure that consists of columns, radial beams, circumferential beams and steel grating and is designed to support the deadweight of non-safety-related equipment and to support safety-related and non-safety-related piping. Design of the DEPSS is to be accomplished in accordance with the ANSI/AISC N-690 Standard. This is acceptable to the staff.

The materials of construction and their fabrication, construction, and installation are in accordance with ACI Standard 349 and ANSI/AISC Standard N-690 for the concrete and steel structures, respectively.

The criteria specified for the analysis, design, construction of the containment internal structures account for anticipated loadings and postulated conditions that may be imposed on the structures during their lifetime conform to the established criteria, codes,

standards, specifications and will be acceptable upon staff's review of information which will be provided in the FSAR.

3.8.4 Other Seismic Category I Structures

Other seismic Category I structures at Lungmen NPS Units 1 and 2 are the Reactor Building (RB) which is integral with the containment structure, the Control Building, the Auxiliary Fuel Building, the Safety Service Water Pump House, and seismic Category I cable trays, conduits, HVAC ducts and their supports. No seismic Category I masonry walls are used in the design. The Radwaste Building is a seismic Category IIB structure. Major structural concrete walls, slabs, columns, and roof of the Radwaste Building are designed to resist the OBE earthquake.

PSAR Section 3.8.4 provides the applicable codes, standards and specifications, loads and load combinations, design and analysis procedures, and structural acceptance criteria for the seismic Category I structures other than the containment and the containment internal structures.

The Lungmen seismic Category I structures are constructed of structural steel or concrete, or both. The structural components consist of slabs, walls, beams, and columns. The major code used in the design of concrete seismic Category I structures is ACI Standard 349 (1990 Edition). For steel seismic Category I structures, ANSI/AISC Standard N-690 (1984 Edition) was used. As discussed in Section 3.8.3 of this report, the use of ACI 349 and ANSI/AISC Standard N-690 for the design of the seismic Category I structures is acceptable to the staff with the exceptions and restrictions discussed in Section 3.8.3 of this report.

The concrete and steel seismic Category I structures are designed and proportioned to resist various combinations of dead loads; live loads; environmental loads, including winds, typhoons, and earthquakes; and loads generated by postulated ruptures of high-energy pipes (such as reaction and jet impingement forces, compartment pressures, and impact effects of whipping pipes). The effects of the hydrodynamic loads on the RB as a result of SRV discharge or LOCA in the containment are accounted for by using the integrated RB and containment shell finite element model in the design analysis. In

response to staff's question on the treatment of dynamic soil pressure induced by the earthquake, TPC responded by stating that the dynamic soil pressure acting on the below grade exterior walls are calculated from the soil-structure interaction analysis. This approach is acceptable.

The design loads, load combinations, load factors, and the associated acceptance criteria for the seismic Category I structures as listed in PSAR Section 3.8.4.3 are generally in compliance with the guidelines of SRP Section 3.8.4.

The design and analysis procedures (PSAR Section 3.8.4.4) used for these seismic Category I structures are acceptable because they are similar to those approved for previous license applications and are in accordance with SRP Section 3.8.4. Analysis and results of seismic Category I structures including safety factors against sliding, overturning, and flotation are to be supplied in FSAR.

In PSAR Section 3.8.4, TPC describes the codes and standards, loads and load combinations, and design procedures and criteria for the design of seismic Category I cable tray, conduit, HVAC and their supports. Transverse and longitudinal bracing are to be provided to various types of frames which form support systems for the cable trays and HVAC ducts to resist seismic loads. For the distributive systems in the Reactor Building, load combinations used in the design analyses include dead load, live load, normal operating load, and dynamic loads of OBE or SSE plus other Reactor Building vibration loads due to SRV or LOCA. PSAR Subsections 3.8.4.4.2 and 3.8.4.4.3 provide two options for designs of cable trays, conduits, HVAC ducts and their supports. One is by one of the analytical methods discussed in PSAR Subsection 3.7.3 and the other is by design-by-rule methods. The staff raised the question of USNRC approval of the design-by-rule methods and their applicability to the Lungmen site. In response, TPC will delete reference of design-by-rule methods in PSAR Subsections 3.8.4.4.2 and 3.8.4.4.3. This is acceptable.

The major codes used for the Radwaste Building are Republic of China Building Code (1997 Edition), ACI 318 (1995 Edition) for concrete structure and AISC (1989 Edition) for structural steel. The building is designed and proportioned to resist various combinations of dead loads; live loads; environmental loads, including winds and seismic. Earthquake loads of either the OBE or ROC Building Code "medium seismicity

zone," whichever is larger will be used for the design seismic loads. The ultimate strength method in ACI 318 is used for design of the reinforced concrete portions. Either the working stress design or the plastic design method based on the AISC is used for the structural steel portions. The design approach for the Radwaste Building is in conformance with Regulatory Guide 1.143. This is acceptable.

The staff noted lack of design commitments for seismic Category I storage tanks (aboveground and/or buried) and underground tunnel and/or piping (e.g., safety service water lines). In response, TPC stated (1) tanks that are located inside seismic Category I buildings will be seismically analyzed per PSAR Section 3.7.3.16 and designed to the ASME III requirements and (2) tanks that are located in the yard will be seismically analyzed per PSAR Section 3.7.3.16 and designed to the requirements to be provided in the FSAR. This is acceptable. However, no descriptions of seismic Category I tanks were provided in TPC's response. Such information is to be provided in the PSAR in order to meet guidelines in Section 3.8.4.1 of RG 1.70, "Standard Format and Content of Safety Analysis Report for Nuclear Power Plants," Revision 3. This is an open item.

The last paragraph of PSAR Section 3.8.4 states that Lungmen Units 1 and 2 do not contain seismic Category I pipeline buried in soil. The staff raised a question concerning design considerations and acceptance criteria of the seismic Category I service water lines. TPC's response to this review question was not responsive. This is an open item.

The applicable codes, standards, and specifications, loads and load combination, design and analysis procedure, and structural acceptance criteria for the Safety Service Water Pump House will be provided in the FSAR, thus was not reviewed by the staff. However, TPC's commitments to the design and construction of the Safety Service Water Pump House should be provided in the PSAR. This is an open item.

The staff noted that there are no subsections in Section 3.8.4 to describe the materials of construction and their fabrication, construction, and installation for the reinforced concrete and structural steel in the RB, control building, auxiliary fuel building and radwaste building substructure. This is an open item.

The Turbine Building is a seismic Category IIA structure as such it must be capable of withstanding the Safe Shutdown Earthquake (SSE) so as not to impair safety function of

the portion of the Main Steam Line (MSL) and condenser housed within the turbine building. Design commitments such as codes and standards, loads and load combinations, structural acceptance criteria, design and analysis procedures, etc. are to be provided in Section 3.8.4 of the PSAR. This is an open item.

The criteria used in the analysis, design, and construction of all Lungmen seismic Category I structures to account for anticipated loadings and postulated conditions that may be imposed upon each structure during its service lifetime are generally in conformance with established criteria, codes, standards, and specifications and will be acceptable upon satisfactory resolutions of open items and staff's review of information to be provided in the FSAR.

3.8.5 Foundations

The Lungmen design employs separate reinforced concrete mat foundations for major seismic Category I structures. The RB foundation, which is integral with the containment foundation, supports the containment structure, reactor pedestal, other internal structures, and the balance of RB structure. Even though the containment structure foundation is integral with the RB foundation, it is a portion of the foundation within the perimeter of the containment structure. Therefore, the foundation was designed as part of the containment boundary. The concrete foundations were designed to resist various combination of dead loads, live loads, environmental loads (including winds, typhoon, OBE and SSE), and loads generated by postulated ruptures of high-energy pipes.

The major code used in the design of concrete mat foundations is the ACI 349 Standard, except for the portion of the foundation within the containment boundary for which the 1989 Edition of ASME Code, Section III, Division 2, was used. The design and analysis procedures, the materials of construction and their fabrication, construction code, and installation used for the seismic Category I foundations, are in accordance with the procedures in ACI 349 and ASME Code, Section III, Division 2. The seismic Category I foundations are designed and proportioned to remain within the limits of these design codes for the applicable load combinations, including those that are considered extreme.

During the PSAR review, the staff noted that foundation waterproofing is used. Staff raised questions regarding seismic shear transfer across the waterproofing membrane. In response TPC stated that (1) the transfer of seismic inertia loads from the foundation to the foundation medium is accounted for using a coupled structure and foundation model in the soil-structure interaction and (2) the effects of waterproofing membrane to the shear transfer are accounted for by using a proper friction coefficient in the foundation sliding evaluation. TPC provided coefficients of friction between various materials such as between mudmat and bedrock/rock fill and between concrete and non-vulcanized butyl rubber. The staff noted that the concrete mat foundation is not in direct contact with the waterproofing membrane due to the presence of the mudmat. Furthermore, the effect of presence of groundwater to the frictional coefficient between the mudmat and the waterproofing membrane needs to be considered. This is an open item.

The criteria used in the analysis, design, and construction of all the plant seismic Category I foundations to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime are in conformance with the established criteria, codes, standards, and specifications.

The staff found TPC's commitments in terms of applicable codes and standards, loads and load combinations, design and analysis procedures, structural acceptance criteria, and materials, quality control, and quality assurance to be acceptable in general. The following open items resulted from the PSAR review of Section 3.8 are to be resolved:

- Track No. 03-028 Question No. 1 and Track No. 03-050 Question No. 1(a), (b), and (c) concerning seismic Category I aboveground and belowgrade storage tanks. See Section 3.8.4 of SER for details of staff's review.

- Track No. 03-028 Question No. 2 concerning seismic Category I safety service water lines. See Section 3.8.4 of SER for details of staff's review.

- Track No. 03-029 Question No. 1 and Track No. 03-051 Question No. 1 concerning shear transfer from mat foundations into foundation media. See Section 3.8.5 of SER for details of staff's review.

- Review question concerning design commitments for the safety service water pump house. See Section 3.8.4 of SER for details of staff's review.

- Review question concerning adding a subsection in Section 3.8.4 on major materials used in the construction of the reactor building, control building, auxiliary fuel building, and radwaste building. See Section 3.8.4 of SER for details of staff's review.

- Review question concerning design of seismic Category IIA turbine building. See Section 3.8.4 of SER for details of staff's review.

4

SER on Lungmen PSAR Chapter 7, "Instrumentation and Control"

7. INSTRUMENTATION AND CONTROL SYSTEMS

The review process, associated review guides, key points for review and review checklists used for the Lungmen Units 1 & 2 PSAR Chapter 7, "Instrumentation and Control System," review were based primarily upon Revision 4 to Chapter 7 of NUREG-0800, the Standard Review Plan (SRP), and 10 CFR 50 (1996 Edition). Revision 4 to Chapter 7 of NUREG-0800, issued in 1997, is the latest issued version of the SRP Chapter 7.

The review process included evaluation of the system concept, system requirements, and system design. The system concept evaluation included the following review points:

- The overall I&C system design's relationship to both the functions required by 10 CFR 50 and the functions required to support the assumptions of the plant accident analysis.
- Compliance with the technically relevant portions of 10 CFR 50.

The system requirements evaluation included the following review points:

- Principal design criteria with respect to the guidance of 10 CFR 50.55a(h) (ANSI/IEEE Std 279) and 10 CFR 50, Appendix A.
- The design bases, including functional design requirements and the relationship of the design bases to the principal design criteria.

The system design evaluation, consistent with the level of detail provided in the PSAR, included the following review points:

- The key characteristics, performance requirements, and general arrangements of the systems to confirm that there is reasonable assurance the final design will conform to the design bases with adequate margin for safety.
- TPC's analysis and technical justification to show that the I&C system design, including the underlying design bases and performance requirements, can perform appropriate safety functions.

Review guidance was based upon and utilized the fundamental acceptance criteria for I&C systems described in 10 CFR 50.55a; ANSI/IEEE Std 279, "Criteria for Protection Systems for Nuclear Power Generating Stations"; Reg. Guide 1.153, "Criteria for Power, Instrumentation, and Control Portions of Safety Systems," and Appendix A of 10 CFR 50 (General Design Criteria). Appendix B of 10 CFR 50 (Quality Assurance Criteria), which provides criteria for quality assurance programs to be applied to the design, fabrication, construction, and testing of I&C safety systems, was also applied. Digital I&C systems were given special attention consistent with NUREG-0800, Chapter 7 guidance and references.

Checklists used in the Chapter 7 review were based upon the review criteria of Table 7-1 of Revision 4 of Chapter 7 of NUREG-0800, "Acceptance Criteria and Guidelines for Instrumentation and Control Systems Important to Safety." This included criteria from:

- 10 CFR Part 50
- General Design Criteria (GDC) of 10 CFR 50 Appendix A
- NRC Staff documents, e.g., SECYs
- Regulatory Guides
- Branch Technical Positions

In addition, IEEE and other standards referenced by the sources listed above were utilized in the review.

Design details (e.g., equipment types and arrangements) for the instrumentation and controls, and in particular the digital equipment are not provided in the PSAR. Accordingly the functional system descriptions, commitments to standards and design processes were used as the bases for an initial determination of the adequacy of the I&C systems.

The findings of the U. S. Nuclear Regulatory Commission (USNRC) in their review of the General Electric Standard Safety Analysis for the Advanced Boiling Water Reactor (ABWR), documented in NUREG 1503 (Reference 3) were also used where applicable.

7.1 Identification of Safety Related Systems and Safety Criteria (PSAR Section 7.0)

7.1.1 Introduction

Chapter 7 of the PSAR presents design and performance information relative to the instrumentation and control (I&C) systems utilized throughout the plant. The design and performance considerations relative to these systems' safety function and their mechanical aspects are described in other PSAR chapters. The instrumentation and control systems are designated as either non-safety-related systems or safety systems, depending on their function. Some portions of a system may have a safety function, while other portions of the same system may be classified non-safety-related.

The systems presented in Chapter 7 are classified according to the NRC Standard Review Plan (SRP) NUREG-0800. Each of the nine categories of I&C systems identified in Chapter 7 of the SRP were reviewed except for the essential auxiliary support systems which are covered in other chapters of the PSAR. These included:

- Reactor trip systems (RTS)
- Engineered safety features actuation systems (ESFAS)
- Safe shutdown systems
- Information systems important to safety
- Interlock systems important to safety
- Control systems
- Diverse instrumentation and control systems
- Data communication systems

Design bases and criteria for I&C systems design are based on the need to have each system perform its intended function while meeting the requirements of applicable general design criteria, regulatory guides, industry standards, and other documents. The safety design basis for a safety system involves the unique design requirements that establish the limits within which the safety objectives shall be met. The general functional requirements portion of the safety design basis consists of those requirements which have been determined to be sufficient to ensure the adequacy and reliability of the system from a safety viewpoint. Many of these requirements are incorporated into various codes, criteria, and regulatory requirements.

Safety systems provide actions necessary to assure safe plant shutdown, to protect the integrity of radioactive material barriers and/or prevent the release of radioactive material in excess of allowable dose limits. These safety systems consist of components, groups of components, systems, or groups of systems. A safety system may have a power generation design basis which states in functional terms the unique design requirements which establish the limits within which the power generation objective for the system shall be set.

Non-safety-related (including power-generation) systems are reactor support systems which are not required to protect the integrity of radioactive material barriers nor prevent the release of radioactive material in excess of allowable dose limits. The I&C portions of these systems may, by their actions, prevent the plant from exceeding preset limits which would otherwise initiate action of the safety systems.

The plant systems have been examined with respect to specific regulatory requirements and industry standards which are applicable to the instrumentation and controls for the various systems. Applicable requirements include specific parts or entities from the following:

1) Title 10 of the U.S. Code of Federal Regulations
2) NRC Regulatory Guides
3) NRC Branch Technical Positions
4) Industry codes and standards

The specific regulatory requirements identified in the Standard Review Plan (SRP) which are applicable to each system's instrumentation and controls are specified in Table 7.1-2 of the PSAR.

7.1.2 Review

The first part of the review focused on identification of safety related systems and the criteria to which these systems conform. Section 7.1 of the Lungmen PSAR defines the I&C systems for the plant and identifies the criteria to which each system conforms. Table 7.1-1 (Table 7.1-2 of the PSAR) provides a matrix of the standards and criteria to which each system conforms. It is formatted in a consistent manner with Table 7.1 of the SRP which identifies standards and criteria to which I&C systems are reviewed for conformance by the USNRC. Comparison of the two tables indicates Table 7.1-2 of the PSAR generally complies with Table 7.1 of the SRP. Deviations of Table 7.1-2 from the SRP are explained in Reference 5. The deviations are considered minor and acceptable.

The second part of the review focused on the concept, requirements, and design of each of the systems listed in 7.1.1. Each of these systems is discussed in Sections 7.2 through 7.9 respectively.

7.1.3 Review Findings

The PSAR has identified the I&C systems which are important to safety in accordance with Reg. Guide 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," November 1978.

The PSAR has identified the acceptance criteria consisting of the General Design Criteria and ANSI/IEEE Std. 279, included in the NRC's regulations, which are applicable to those systems as identified in the SRP. The PSAR has also identified the guidelines consisting of the regulatory guides and the industry codes and standards which are applicable to the systems as identified in the SRP. The review concludes that the implementation of the identified acceptance criteria and guidelines will satisfy the requirements of GDC 1 with respect to designing to quality standards commensurate with the importance of the safety functions to be performed.

7.2 Reactor Trip System - Instrumentation and Control (PSAR Section 7.2)

7.2.1 Introduction

Table 7.1-1.
Regulatory Requirements Applicability Matrix for I&C systems

(Table 7.1-2 of PSAR)

Table 7.1-1. (cont)
Regulatory Requirements Applicability Matrix for I&C Systems (Continued)

(Table 7.1-2 of PSAR cont)

The review covered the reactor trip system (RTS), which is part of the reactor protection system, and includes all equipment from sensors to actuation devices that are required to initiate reactor shutdown. The RTS automatically initiates the reactivity control system (control rods) to ensure that specified acceptable fuel design limits are not exceeded.

The objectives of the RTS review were to confirm that the RTS design (1) satisfies the requirements of the acceptance criteria and guidelines applicable to the protection system, and (2) will perform its safety functions for all plant conditions under which they are required.

The acceptance criteria and guidelines acceptable to the RTS identified in the SRP Section 7.2 were applied.

7.2.2 Reactor Trip System Description

The reactor trip system for the Lungmen design is called the Reactor Protection System (RPS). It is described in Section 7.2 of the PSAR. The RPS is the overall complex of instrument channels, trip logics, trip actuators and scram logic circuitry that initiate rapid insertion of control rods (scram) to shut down the reactor. The RPS also establishes reactor operating modes and provides status and control signals to other systems and annunciators. The decision-making segment of the RPS is the Safety System Logic and Control (SSLC) system which is discussed in Section 7.2.3.

The Reactor Protection System (RPS) and its supporting instrumentation and control system apparatus, the Safety System Logic and Control (SSLC), are designed to meet the following functional requirements:

1. Initiate a reactor scram with precision and reliability to prevent or limit fuel damage following abnormal operational transients.
2. Initiate a scram with precision and reliability to prevent damage to the reactor coolant pressure boundary as a result of excessive internal pressure (i.e., to prevent nuclear system pressure from exceeding the limit allowed by applicable industry codes).
3. Limit the uncontrolled release of radioactive materials from the fuel assembly or reactor coolant pressure boundary, by precisely and reliably initiating a reactor scram on gross failure of either of these barriers.
4. Detect conditions that threaten the fuel assembly or reactor coolant pressure boundary from inputs derived from variables that are true, direct measures of operational conditions.
5. Respond correctly to the sensed variables over the expected range of magnitudes and rates of change.

6. Provide a sufficient number of sensors for monitoring essential variables that have spatial dependence.

The following design bases assure RPS/SSLC reliability:

7. If a single random failure can cause a control system action that causes a plant condition that requires a reactor scram but also prevention by some RPS channels, the remaining portions of the RPS/SSLC shall meet the functional requirements (items a, b, and c above), even when degraded by a second random failure.
8. Loss of one power supply shall neither directly cause nor prevent a reactor scram.
9. Once initiated, an RPS action shall go to completion. Return to normal operation shall require deliberate operator action.
10. There shall be sufficient electrical and physical separation between redundant I&C equipment monitoring the same variable to prevent environmental factors, electrical transients, or physical events from impairing the ability of the system to respond correctly.
11. No single failure within the RPS/SSLC shall prevent proper RPS action when required to satisfy Safety Design Bases as described by a, b, and c above.
12. Any one intentional bypass, maintenance operation, calibration operation, or test to verify operational availability shall not prevent the ability of the reactor protection system to respond correctly.
13. The system shall be designed so that two or more sensors for any monitored variable exceeding the scram setpoint will initiate an automatic scram.

The following bases reduce the probability that RPS/SSLC operational reliability and precision will be degraded by operator error:

14. Access to trip settings, component calibration controls, test points, and other terminal points shall be under the control of plant operations supervisory personnel.
15. Manual bypass of instrumentation and control equipment components shall be under the control of the MCR operator. If the ability to trip some essential part of the system has been bypassed, this fact shall be continuously annunciated in the MCR.
16. Selective automatic and manual operational trip bypasses are provided as necessary, to permit proper plant operation. Those bypasses allow meeting protection requirements that depend upon specific existing or subsequent reactor operating conditions.
17. Manual control switches are provided for initiation of reactor scram by plant operator when necessary.
18. Mode selection is provided for enabling the appropriate instrument channel trip functions required in a particular mode of operation.

Specific requirements applicable to the instrumentation and control portion of RPS are shown in Table 7.1-1 (Table 7.1-2 of the PSAR).

The RPS/SSLC is designed with the added objective of plant availability. The setpoints, power sources, and instrumentation and controls are to be arranged in such a manner as to preclude spurious scrams insofar as practicable and safe.

The RPS will initiate a reactor scram when any one or more of the following conditions occur or exist within the plant:

1) Neutron Monitoring System monitored conditions exceed acceptable limits
2) High Reactor Pressure
3) Low Reactor Water Level (Level 3)
4) High Drywell Pressure
5) Main Steamline Isolation Valve Closure (Inboard or Outboard)
6) Low Control Rod Drive HCU Accumulator Charging Header Pressure
7) Turbine Stop Valve Closed
8) Turbine Control Valve Fast Closure
9) High Suppression Pool Temperature
10) High Seismic Acceleration in the Reactor Building
11) Operator-initiated Manual Scram

The combination of division trip, scram, reset and bypass logic that make up the overall RPS logic is shown in Figure 7.2-1 (Figure 7.2-1 of the PSAR) and Figure 7.2-2 (Figure 7.2-2 of the PSAR). Each division trip logic receives trip inputs from all four sensor channels and Neutron Monitoring System (NMS) divisions and provides a sealed-in trip output to the scram logic when the same trip condition exists in any two or more sensor channels or NMS divisions. At the division trip logic level, various trips and trip initiating conditions can be bypassed. The scram logic will initiate a reactor scram when a trip condition exists in any two or more division trip logics. At the scram logic level, no bypasses are possible.

The redundant portions of the RPS have physically separated sensor taps, sensing lines, sensors, sensor rack locations, cable routing, and termination in four separate SSLC panels in the MCR. By the use of four or more separate redundant sensors for each RPS variable with separate redundant logic and wiring, the RPS has been protected from a credible single failure. Redundance of the RPS logic power supply is also provided. There are four Class 1E uninterruptible power sources and four Class 1E DC power sources which supply electrical power, one set to each division

(Figure 7.2-1 of PSAR)

Figure 7.2-1. RPS Equipment Arrangement

of the RPS. A loss of one power supply will neither inhibit protective action nor cause a scram.

Four independent sensor channels monitor the various process variables. The redundant sensor devices are separated so that no single failure can prevent a scram. Divisional separation is also applied to the Essential Multiplexing System (EMS), which provides data highways for the sensor input to the logic units. The EMS is discussed in Section 7.9. The four divisions of RPS scram logic reside in four physically separated SSLC cabinets. SSLC is discussed in Section 7.2.3. Fiber-optic cable is used from remote multiplexing units (RMUs) of EMS to control room equipment.

Virtually all hardware within the RPS design scope is located within the four separate and redundant safety system logic and control (SSLC) cabinets in the MCR area, except for the instrumentation monitoring turbine stop valve closure, turbine control valve fast closure, and turbine bypass valve position. The equipment for initiating RPS scram time testing is also located outside the MCR area. The SSLC panels are mounted in the divisional back panel areas of the MCR. The major control switches are located on the main control console.

The SSLC logic cabinets, which contain the RPS functions for Divisions, I, II, III, and IV, include a separate cabinet for each division. The cabinets contain digital and solid-state discrete and integrated circuits used to condition signals transferred to the SSLC from EMS. They also contain combinational and sequential logic circuits for the initiation of safety actions and/or alarm annunciation, isolators for electrical and physical separation of circuits used to transmit signals between redundant safety systems or between safety and non-safety systems, and system support circuits such as power supplies, automatic testing circuits, etc. Load drivers with solid-state switching outputs for actuation of solenoids are located in the MCR.

The RPS is classified as Safety Class 2, Seismic Category I, and Quality Group B (electric Safety Class 1E) per Regulatory Guide 1.26 and meets the requirements of 10 CFR 50.55a(h). It utilizes two types of power: (1) 120 V vital AC - taken from the four divisional SSLC power supply buses and (2) 125VDC - taken from two of the four divisional SSLC battery buses.

The RPS is designed to provide reliable single-failure-proof capability to automatically or manually initiate a reactor scram while maintaining protection against unnecessary scrams resulting from single failures. This is accomplished through the combination of fail-safe equipment design and redundant two-out-of-four logic, arrangement. All equipment supporting RPS is designed to fail into a trip initiating state on loss of power or input signal. In conjunction with the trip

initiating logic signals to and within the RPS are asserted low, whereas trip bypass logic signals and trip bypass permissive logic, signals are asserted high.

(Figure 7.2.2 of PSAR)

Figure 7.2-2. RPS Control Interface Diagram

The RPS equipment is divided into four redundant instrument channels (channels of sensors), four divisions of trip logic, four divisions of trip actuators, and two divisions of manual scram controls and scram logic circuitry. The instrument channels, divisions of trip logics, divisions of trip actuators and associated portions of the divisions of scram logic circuitry together constitute the RPS automatic scram and backup scram manual initiation logic. The automatic and manual scram initiation logics are independent of each other.

7.2.3 Safety System Logic and Control (SSLC) Description

The integrated digital protection system, designated as safety system logic and control (SSLC), is the decision-making segment of RPS and ESF, and provides their supporting structure (physical and functional). SSLC processes automatic and manual demands for reactor trip, ESF system initiation, or containment isolation based upon sensed plant process parameters or operator request. SSLC runs without interruption in all modes of plant operation to support the required safety functions.

The SSLC multi-divisional arrangement includes divisionally separate control room and other panels which house the SSLC equipment for controlling the various safety function actuation devices. The SSLC receives input signals from the redundant channels of safety-related instrumentation, and uses the input information to perform logic functions in making decisions for safety actions. Figure 7.2-3 (Figure 7.1-1 of the PSAR) illustrates the SSLC arrangement.

Most SSLC input data is multiplexed from process variables monitored by the essential multiplexing system (EMS) (refer to Section 7.9) in four physically and electrically isolated instrumentation divisions. Each of the four independent and separated EMS channels feeds an independent and separated channel of SSLC equipment in the same division via remote multiplexing units (RMUs) that furnish plant sensor data to the fiber optic EMS network.

Signals without access to EMS, signals that must meet time response constraints, and signals from system logic that is in close proximity to the SSLC cabinets are directly connected to the four divisional cabinets in the main control room panel (MCRP) area. All signals to SSLC are derived from sensors that are redundant in the four divisions for each sensed variable. All input data, whether originally in analog form or discrete (contact closure) form, is converted and processed as serial-format digital bit streams within EMS and SSLC. RPS output signals in a division are sent to equipment actuators in that division by being hardwired directly as bistable trips. Alarm and status signals are sent from SSLC to the control room data network for computer logging and display.

(Figure 7.1-1 of PSAR)

Figure 7.2-3. Assignment of Interfacing Safety System Logic to SSLC Controllers

The ESF output channels, which initiate slow-responding mechanical devices, are implemented via EMS. Control output signals in a division are sent to equipment actuators in that division by being multiplexed as digital messages to local Class 1E RMUs and then converted to bistable trips. A confirmed trip actuates a solid-state power switch, designated as a load driver (LD), at a Remote Multiplexing Unit (RMU) of EMS located in the control building (CB), reactor building (RB), or safety-related circulating water pumphouse. The trip is hardwired from the RMU to the equipment actuator (e.g., pump motor or motor-operated valve). ESF output channels reside only in divisions I, II, and III because ESF mechanical equipment is in these three divisions and not in Division IV.

SSLC does not require operator intervention during normal operation. However, abnormal conditions or maintenance activities require the operator to manually bypass a division of sensor inputs or a division of RPS trip logic. Since ESF trips are processed in dual-redundant channels of ESF outputs within a division, these trips are automatically bypassed and switched to the redundant backup channel on a sensed failure via self-test (manual output bypass is also available). Failure of automatic logic requires the appropriate emergency manual response (e.g., initiation of an emergency core cooling system or PCV isolation). However, all ESF loops have diverse or redundant backup loops in separate and independent mechanical and electrical divisions, so emergency manual initiation should not be required.

Manual control switches for reactor (trip) scram and main steamline isolation valve closure are independent of microprocessor-controlled logic. The two-button manual scram (and scram reset) of RPS is also independent of all logic that controls the loads of the RPS scram pilot valve solenoids. Manual scram is performed by opening the power sources to the solenoids. Control switches for ECCS operate through dual-redundant microprocessor logic in each of the three ECCS divisions.

SSLC components, as the supporting structure of RPS and ESF, are classified as Safety Class 2, Seismic Category I, and Quality Group B (Electric Safety Class 1E) per Regulatory Guide 1.26 and meet the requirements of 10CFR50.55a(h).

To support the safety-related trip requirements of its interfacing systems, SSLC uses two types of power sources. Both sources are used simultaneously for redundant backup of the logic processors, which contain dual-redundant low voltage power supplies. The sources are (1) Class 1E 120 V vital AC (uninterruptible) taken from the four divisional SSLC power supply buses and (2) Class 1E 125 VDC taken from the four divisional SSLC battery buses.

Software for SSLC operation is to be developed or purchased and qualified for safety-related use together with the microprocessor-based controller hardware

according to the software management plans described in Section 7.1.1.2.1.3 of the PSAR. Software-based control programs are embedded as firmware in the controller hardware and cannot be changed during operation.

The requirements to be met by the hardware and software development implementation activities for safety-related systems supported by SSLC are defined in Section 7.1.1.2.1.3 of the PSAR. A Software Management Plan, Configuration Management Plan, and Verification and Validation Plan are to be generated to conform to the guidance of BTP-HICB-14.

The SSLC logic cabinets, which contain the RPS and ESF functions for Divisions I, II, III, and IV, include a separate cabinet for each division. The cabinets contain digital and solid-state discrete and integrated circuits used to condition signals transferred to the SSLC from EMS. They also contain combinational and sequential logic circuits for the initiation of safety actions and/or alarm annunciation, isolators for electrical and physical separation of circuits used to transmit signals between redundant safety systems or between safety and non-safety systems, and system support circuits such as power supplies, automatic testing circuits, etc. Most logic functions are implemented in software permanently embedded in digital memory. However, some functions remain hardwired for reasons of diversity or speed of response. Load drivers with solid-state switching outputs for actuation of solenoids, motor control centers, or switchgear are located in the control room area in a separate bay of the SSLC cabinets.

7.2.4 Review Findings

The review concludes that the design of the reactor trip system (RTS) and the RTS initiation of the essential auxiliary support (EAS) systems are acceptable and meet the relevant requirements of General Design Criteria (GDC) 1, 2, 4, 13, 19-25, and 29, and 10CRF50.34(f), 50.55a(a)(1), and 50.55a(h).

The review of these systems was for conformance to the guidelines in the regulatory guides and industry codes and standards applicable to these systems. The review concludes that the PSAR adequately identifies the guidelines applicable to these systems. Based upon the review of the system design for conformance to the guidelines, there is reasonable assurance that the systems fully conform to the guidelines applicable to these systems. Therefore the requirements of GDC 1 and 10 CFR 50.55a(a)(1) are met by the design.

The review included the identification of those systems and components for the RTS designed to survive the effects of earthquakes, other natural phenomena, abnormal environments and missiles. The review concludes that the PSAR has identified

those systems and components consistent with the design bases for those systems. Therefore the review finds that the identification of these systems and components satisfies the requirements of GDC 2 and 4.

Based on the review of RTS system status information, manual initiation capabilities, and provisions to support safe shutdown, the review concludes that information is provided to monitor the RTS over the anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety. Appropriate controls are provided for manual initiation of reactor trip. The RTS appropriately supports actions to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. Therefore, the review finds that the RTS design satisfies the requirements of GDC 13 and 19.

The review of system functions concludes that the RTS conforms to the design bases requirements of ANSI/IEEE Std 279 and 10CFR50.34(f). The RTS conforms to the guidance of Reg. Guide 1.105. The review concludes that the RTS includes the provision to sense accident conditions and anticipated operational occurrences in order to initiate reactor shutdown consistent with the accident analysis presented in Chapter 15 of the SAR and evaluated in the SER. Therefore the review finds that the RTS satisfies the requirements of GDC 20.

The RTS conforms to the guidelines for periodic testing in Reg. Guide 1.22 and Reg. Guide 1.118. The bypassed and inoperable status indication conforms to the guidelines of Reg. Guide 1.47. The RTS conforms to the guidelines on the application of the single-failure criterion in ANSI/IEEE Std 379 as supplemented by Reg. Guide 1.53. The review concludes that the RTS satisfies the requirement of ANSI/IEEE Std 279 with regard to the system reliability and testability. Therefore the review finds that the RTS satisfies these requirements of GDC 21.

The RTS conforms to the guidelines in Reg. Guide 1.75 for the protection system independence. The review concludes that the RTS satisfies the requirement of ANSI/IEEE Std 279 with regard to the systems independence. Therefore, the review finds that the RTS satisfies the requirements of GDC 22.

The review of the RTS concludes that the system is designed to fail into a safe mode if conditions such as disconnection of the system, loss of energy, or postulated adverse environment are experienced. Therefore, the review finds that the RTS satisfies the requirements of GDC 23.

Based on the review of the interfaces between the RTS and plant operating control systems, the review concludes that the system satisfies the requirements of ANSI/IEEE Std 279 with regard to control and protection system interactions. Therefore the review finds the RTS satisfies the requirements of GDC 24.

The review of the RTS concludes that the system satisfies the protection system requirements for malfunctions of the reactivity control system such as accidental withdrawal of control rods. Therefore, the review finds that the RTS satisfies the requirements of GDC 25.

The review of all the above GDCs concludes that the RTS satisfies the requirements of GDC 29.

The review conclusions noted above are based upon the requirements of ANSI/IEEE Std 279 with respect to the design of the RTS. Therefore the review finds that the RTS satisfies the requirement of 10CRF50.55a(h) with regard to ANSI/IEEE Std 279.

The review of the RTS examined the dependence of this system on the availability of essential auxiliary systems. The review concludes that the design of the RTS is compatible with the functional requirements of EAS systems.

Based on the review of software development plans and PSAR commitments to guidance documents, the review concludes that the computer systems design meets the guidance of Reg. Guide 1.152. In particular, to comply with the regulatory criteria and standards, the digital protection system was designed with the following features:

- Four independent and separated equipment divisions
- Segmentation of functions within divisions to improve defense-in-depth
- Separate controllers with limited functions (setpoint determination, interlock logic, trip seal-in)
- Each controller is a self-contained logic processor with built-in self-test
- Software embedded in programmable read-only memory (PROM) which cannot be changed by operator or external signals
- Fiber-optic transmission media between safety divisions and from safety to non-safety divisions
- Fault-tolerant (dual-redundant per division) and automatically reconfigurable after single failure multiplexer
- Continuous self-test and fault localization
- Two-out-of-four coincidence voting in all four divisions.
- Process control functions are contained in separate controllers with a separate diverse triplicated multiplexing system
- 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 within each division of functions that are performed with the reactor at high pressure and functions that are performed with the reactor at low pressure.

- The design process includes a software QA plan, hardware and software V&V, the use of software tools, the requirements to demonstrate system integrity, designing for test and calibration, methods for determining the need for diversity, and qualification of commercial computers. It also includes a lifecycle with the following categories:
  1) planning
  2) design definition
  3) software design
  4) software coding
  5) integration
  6) validation
  7) change control

  It further includes the following documents which are typical of a software development program.

  1) The software management plan (SMP) establishes the organization and authority structure for the design, the procedures to be used, and the interrelationships between major activities.
  2) The software configuration management plan (CMP) provides the means to identify software products, control and implement changes, and record and report change implementation status.
  3) The verification and validation (V&V) plan describes the method to ensure that the requirements of each phase or stage are fully and accurately implemented in the next phase.

  Having these programs and features in place permits detailed design to proceed with confidence that all regulations will be fulfilled.

The following review topics were considered for the digital systems:

- Equipment qualification, including electromagnetic compatibility
- Real-time, deterministic performance
- On-line and periodic test provisions
- Communications independence
- Control of access

Additional discussion of the digital protection system compliance with R. G. 1.152 and ANSI/IEEE 7-4.3.2-1993 is provided in Reference 4.

Therefore, the special characteristics of computer systems are adequately addressed, and the review finds that the RTS satisfies the requirements of GDC 1 and 21.

Based on the review of the PSAR commitments to defense-in-depth and diversity the review concludes that the RTS complies with the criteria for defense against common-mode failure in digital instrumentation and control systems. Therefore, the review finds that adequate diversity and defense against common-mode failure has been provided to satisfy these requirements of GDC 21 and 22, and Item II.Q of the Staff Requirements Memorandum on SECY-93-087.

7.3 Engineered Safety Features Systems Instrumentation and Control (PSAR Section 7.3)

7.3.1 Introduction

The objective of the review of the Engineered Safety Features (ESF) systems was to confirm that the ESF control system and ESF actuation systems (ESFAS) satisfy regulatory acceptance criteria, guidelines, and performance requirements. Acceptance criteria and guidelines applicable to the ESFAS and ESF control systems identified in SRP Section 7.3 were used in the review.

7.3.1 Engineered Safety Feature Systems Descriptions

Section 7.3 of the PSAR describes the instrumentation and controls for the various engineered safety features (ESF) systems and provides design basis information called for by IEEE 279.

The control logic for the ESF systems is part of the Safety System Logic and Control (SSLC) discussed in Section 7.2. Process sensors critical for initiation of these systems are redundant in the four independent and separated divisions of SSLC instrument channels. These sensor signals are transmitted over the Essential Multiplexing System (EMS) from local plant areas to the main control room (MCR) location of the four SSLC cabinets. The resulting ESF command signals are transmitted by EMS to the remote multiplexing units in the Reactor Building, where they are distributed to the ESF equipment drivers (e.g., motor control centers and switchgear).

The ESF systems include the following:

1) Emergency Core Cooling Systems (ECCS)
2) Leak Detection and Isolation System (LDI)
3) Wetwell and Drywell Spray Mode of the Residual Heat Removal System (RHR) (WDCS-RHR)
4) Suppression Pool Cooling Mode of the RHR System (SPC-RHR)
5) Standby Gas Treatment System (SGT)
6) Emergency Diesel Generator (EDG) support systems

7) Reactor Building Cooling Water (RBCW) System and Reactor Building Service Water (RBSW) System
8) Essential HVAC Systems
9) Emergency Chilled Water System (ECW)
10) Nitrogen Supply System (N_2)
11) Flammability Control System (FCS)

7.3.1.1 Emergency Core Cooling Systems Instrumentation and Controls

The Emergency Core Cooling Systems (ECCS) consist of the four systems discussed in this section. The purpose of ECCS instrumentation and controls is to sense the need for ECCS action and to initiate appropriate response from the system in the event of an accident requiring its action. The ECCS instrument channels detect a need for core cooling systems operation, the logic makes appropriate decisions, and the trip actuators initiate the appropriate equipment operation.

1) High Pressure Core Flooder (HPCF) System

The I&C components for the HPCF System, except as noted in the PSAR, are located outside the drywell. Pressure and level transducers used for HPCF initiation are part of the Main Steam (MS) and Containment Monitoring (CMS) Systems and are located on racks outside the drywell. The system is arranged to allow a design flow functional test during normal reactor power operation. Supporting systems for the HPCF I&C consist only of the instrumentation, logic, and motive power supplies. The HPCF System is designed to operate from preferred offsite power sources or from the Division 2 and 3 diesel generators if offsite (preferred) power is not available.

2) Automatic Depressurization Subsystem (ADS) (SRV electrical activation logic)

Automatic depressurization by the Automatic Depressurization System (ADS) (part of MS) is provided to reduce the pressure during a loss-of-coolant accident event in which the HPCF and RCIC are unable to provide sufficient water to restore vessel water level. Reduction in reactor pressure allows additional cooling water by the low pressure makeup system (RHR/LP flooder mode). Reduced reactor pressure also allows the HPCF loops to be able to provide more water than they were capable of at higher reactor pressure. Automatic safety/relief valves (SRVs) are installed on the main steam lines inside the drywell. Eight of the eighteen SRVs are designated as ADS valves and are capable of operating from either ADS logic or safety/relief logic signals. These SRVs can be actuated in two ways: (1) by pneumatic actuation with electrical control power or (2) by mechanical actuation using inlet steam pressure at predetermined setpoint. The suppression pool provides a heat sink for steam relieved by these valves. Relief (power actuated) valve operation may be controlled manually from the control

room. Supporting systems for the ADS include the instrumentation, logic, control and motive power sources.

The ADS controls consist of redundant trip channels arranged in two separate logics that control two separate solenoid-operated gas pilot valves on each ADS valve. Either pilot valve can operate its associated ADS valve. These pilot valves control the pneumatic pressure applied by accumulators and the N_2. The operator can also control the SRVs manually. Separate accumulators are provided with the control equipment to store pneumatic energy for relief valve operation.

The ADS accumulators are sized to operate the SRV one time at drywell design pressure or five times at drywell normal pressure, following failure of the pneumatic (charging) supply to the accumulator. The pressure sensors provide inputs to local multiplexer units which perform signal conditioning and analog-to-digital conversion. The formatted, digitized sensor inputs are multiplexed with other sensor signals over optical data links to the logic processing units in the MCR. All four transmitter signals are fed into the two-out-of four logic for each of the two divisions, either of which can actuate the ADS. Station batteries and SSLC power supplies energize the electrical control circuitry. The power supplies for the redundant divisions are separated to limit the effects of electrical failures. Electrical elements in the control system energize to cause the relief valves to open.

3) Reactor Core Isolation Cooling (RCIC) System

The instrumentation and controls (I&C) for the Reactor Core Isolation Cooling (RCIC) System provide control for the pump/turbine valves, and accessories to assure that sufficient reactor water inventory is maintained in the reactor vessel to permit adequate core cooling to take place for:

a. A small line break loss-of-coolant accident (LOCA) event.
b. When the reactor vessel is isolated and yet maintained in the hot standby condition.
c. When the reactor vessel is isolated and accompanied by a loss of normal coolant flow from the reactor feedwater system.
d. When a complete plant shutdown under conditions of loss of normal feedwater is started before the reactor is depressurized sufficiently for the reactor shutdown cooling mode of the RHR System to be placed into operation.
e. Should a complete loss of AC power occur, the RCIC System is designed to operate for at least 8 hours for these conditions.

The RCIC is classified as a safety-related system powered by the Division I 125 VDC system, except for the steam supply isolation valves. The inboard steam

isolation valves are powered by 480 VAC Division I and the outboard steam supply isolation valve is powered by 125 VDC Division II.

When actuated, the RCIC System pumps demineralized water from the condensate storage tank to the reactor vessel. The suppression pool provides an alternate source of water. The RCIC System includes a 100% capacity steam-driven turbine which drives a 100% capacity pump assembly, turbine and pump accessories, piping, valves, and instrumentation necessary to implement several flow paths.

Level transducers used for the initiation and stopping of RCIC are provided by the MS and are shared by other system channels within each division. High drywell pressure signals are provided by the Containment Monitoring System and are also shared by other system channels within each division. These are located on instrument panels outside the drywell but inside the containment. The only operating components of RCIC that are located inside the drywell are the inboard steamline isolation valve and the steamline warmup line isolation valve. The rest of RCIC normal I&C components are located in the Reactor Building. Cables connect the sensors (via EMS multiplexed optical data links) to control circuitry in the MCR.

4) Low-Pressure Flooder (LPFL) mode of the Residual Heat Removal (RHR) System.

The Low Pressure Flooder (LPFL) Subsystem is an operating mode of the Residual Heat Removal (RHR) System. The LPFL Subsystem is designed to provide water to the reactor vessel following the design basis LOCA. Supporting systems for the LPFL Subsystem include only the instrumentation, control and motive power supplies. Divisions I, II, and III are used for the three loops of the LPFL. The RHR System includes the equipment used for LPFL operation. Control and instrumentation required for the operation of the LPFL mode are safety-related. The instrumentation for LPFL operation controls all necessary valves in the RHR System. This ensures that the water pumped from the suppression pool by the main system pumps is routed directly to the reactor. LPFL operation uses three pump loops, each loop with its own separate vessel injection path. Except for the shutdown cooling inboard suction isolation valves and the testable check valves for Divisions II and III, the components pertinent to LPFL operation are located outside the drywell. Motive power for the RHR System pumps is supplied from AC buses that can receive standby AC power. The three pumps are powered from Division I, II, and III ESF buses, which also provide power to the RCIC (Division I) and HPCF (Divisions II and III) Systems. Motive power for the automatic valves comes from the bus that powers the pumps for that division, except for special cases.

The instrumentation and controls portion of ECCS (i.e., SSLC) is designed to meet the following requirements:

1. Automatically initiate and control the Emergency Core Cooling System (ECCS) to prevent fuel cladding temperatures from reaching the limits of 10CRF50.46.
2. Respond to a need for emergency core cooling regardless of the physical location of the malfunction or break that causes the need.
3. Limit dependence on operator judgment in times of stress by:
   i. Automatic response of the ECCS so that no action is required of plant operators within 30 minutes after a loss-of-coolant accident.
   ii. Indication of performance of the ECCS by MCR instrumentation.
   iii. Provision for manual control of the ECCS in the MCR.

Specific regulatory requirements applicable to the instrumentation and controls for the ECCS are shown in Table 7.1-1 (Table 7.1-2 of the PSAR).

7.3.1.2 Leak Detection and Isolation System (LDI) I&C Description

The LDI is an instrument and control system that utilizes temperature, pressure, radiation and flow sensor signals to detect, indicate and alarm leakage from the reactor coolant pressure boundary (RCPB). The sensors are part of each individual system that provides inputs to LDI. LDI logic operations are performed by the safety system logic and control system (SSLC). SSLC contains the software necessary to perform the logic functions identified as Leak Detection and Isolation. LDI resides as a series of logic functions within the SSLC and does not contain any additional hardware of its own.

LDI transmits signals from other instrumentation systems to SSLC to automatically isolate reactor coolant leaks external to the containment and prevent unacceptable radiological releases from the containment following detection of leakage within the containment. Manual isolation control switches are provided to permit the operator to manually initiate (at the system level) isolation from the main control room (MCR). In addition, each power-operated isolation valve is provided with a separate manual control switch in the MCR which is independent of the automatic and manual leak detection isolation logic.

All LDI logic power is supplied by the respective divisional SSLC logic power supplies. The power for the MSIVs pilot solenoid valve control logic is supplied from all four divisions of the SSLC buses. The MSIVs are spring-loaded, piston-operated pneumatic valves designed to fail closed on loss of electric power or pressure to the valve operator.

The motor-operated isolation shutdown cooling valves in the RHR shutdown cooling loop are isolated by power supplied from divisional power sources. RHR inboard valves are isolated by Division I logic for RHR A, by Division II logic for

RHR B, and by Division III logic for RHR C. RHR outboard valves are isolated by Division II logic for RHR A, by Division III logic for RHR B, and by Division I logic for RHR C. RCIC inboard valves are isolated by Division I logic. RCIC outboard valves are isolated by Division II logic.

The general functional requirements of the Leak Detection and Isolation System (LDI) instrumentation and controls are to detect, indicate and alarm leakage from the reactor primary pressure boundary and, in certain cases, to initiate closure of isolation valves to shut off leakage external to the containment.

In order to meet the safety design basis, the LDI system is designed to:

1. Provide direct and accurate measurements of parameters which are indicative of a reactor coolant pressure boundary (RCPB) leak or a leak of reactor coolant outside the containment and then provide automatic isolation of the affected system or area.
2. Monitor predetermined parameters with precision and reliability and respond correctly to the sensed parameters.
3. Provide a sufficient number of independent monitors, sensing each parameter to ensure accurate measurement and preclude the possibility of a failure to isolate due to instrumentation failure.
4. Provide an isolation control system with sufficient redundancy to ensure that the LDI can perform its intended function, assuming a single failure caused by any of the design basis events or a single power supply failure.
5. Provide an isolation control system which will ensure that isolation of the containment and/or reactor vessel will occur once initiated.
6. Provide instrumentation and control to permit the operator to manually initiate isolation if necessary.
7. Provide interlocks to assure reset capability is only possible after clearance of isolation signals.

Specific regulatory requirements applicable to this system are shown in Table 7.1-1 (Table 7.1-2 of the PSAR).

7.3.1.3 RHR/Wetwell and Drywell Spray Cooling Mode Instrumentation and Controls

Wetwell/drywell spray cooling (WDSC) is a manually initiated operating mode of RHR. It is designed to provide the capability of condensing steam in the wetwell and drywell air volumes. This capability is only available in RHR loops B and C. Power for RHR pumps B and C is supplied from two independent AC buses that can receive standby AC power. Motive and control power for the two divisions of WDSC and I&C equipment are the same as those used for LPFL B and C, respectively.

The wetwell and drywell cooling mode of the RHR provide instrumentation and controls within SSLC that is designed to the following functional requirements:

1. Initiate wetwell and drywell spray as required to avoid environmental conditions of pressure and temperature that would threaten the integrity of the containment during a transient or accident condition.
2. Provide information (from wetwell and drywell pressure sensors) to the operator for manual system initiation in order to provide condensation of steam in the wetwell and drywell air volumes during a transient or accident event.
3. Manually control the wetwell and drywell spray from the MCR.
4. Indicate performance of the wetwell and drywell spray from the MCR

Specific regulatory requirements applicable to the containment spray system are listed in Table 7.1-1 (Table 7.1-2 of the PSAR).

7.3.1.4 RHR/Suppression Pool Cooling Mode Instrumentation and Controls

Suppression pool cooling is an operating mode of the RHR System. It is designed to provide the capability of removing heat from the suppression pool water. The system is automatically initiated upon receipt of a high temperature signal monitored by CMS. The RHR mode may be manually initiated when necessary.

Power for RHR System pumps A, B, and C is supplied from three independent AC buses that can receive standby AC power. Motive and control power for the three loops of suppression pool cooling instrumentation and control equipment are the same as that used for LPFL, A, B, and C, respectively. Suppression Pool cooling (SPC) uses three pump loops, each loop with its own separate discharge valve.

Control and instrumentation for the following equipment is required for this mode of operation:

- Three RHR main system pumps (one in each RHR loop)
- Pump suction valves
- Suppression pool discharge valves

All I&C components pertinent to suppression pool cooling operation, except suppression pool temperature monitoring, are located outside of the drywell.

The Suppression Pool Cooling (SPC) mode is automatically initiated on high suppression pool temperature or manually initiated from the MCR. This mode is put into operation to limit the water temperature in the suppression pool such that the temperature immediately after a blowdown does not exceed the established limit when reactor pressure is above the limit for cold shutdown

The instrumentation and controls of the SPC mode are designed to provide automatic initiation of suppression pool cooling upon sensed high temperature in the suppression pool. The reactor operator may also manually initiate suppression pool cooling to ensure that the pool temperature does not exceed the preestablished pool temperature.

Specific regulatory requirements are listed in Table 7.1-1 (Table 7.1-2 of the PSAR).

7.3.1.5 Standby Gas Treatment System Instrumentation and Controls

The Standby Gas Treatment (SGT) Processes gaseous effluent from the primary and secondary containments when required to limit the discharge of radioactivity to the environment during normal and abnormal operation. It also controls the exfiltration of fission products by maintaining a negative pressure in the secondary containment, and by filtering the effluent prior to discharge to the atmosphere following a LOCA or fuel handling accident. The instrumentation and controls of the SGT are supplied by Class 1E power Division II and Division III.

Process gas flow is controlled manually by a motor-driven butterfly valve located upstream of the filter train. The relative humidity of the air entering the charcoal adsorber is sensed by a humidity element downstream of the electric space heaters. A controller operates the space heaters to maintain the relative humidity of the air at 70% or less. The switch initiates an alarm in the MCR upon high air temperature. Temperature sensors determine the charcoal bed temperature. A switch actuates a control room annunciator upon high temperature in the charcoal.

The instrumentation and controls of this system are designed such that it maintains a negative pressure in the secondary containment, relative to the outdoor atmosphere, in order to control exfiltration of fission products after either (a) a loss-of-coolant accident (LOCA) or (b) a high level of radioactivity in the secondary containment exhaust. The system also is required to filter airborne radioactivity (particulate and halogen) in the effluent to reduce post-accident offsite exposure.

The specific regulatory requirements applicable to this system are given in Table 7.1-1 (Table 7.1-2 of the PSAR).

7.3.1.6 Emergency Diesel Generator (EDG) Support Systems

Emergency diesel generator system (EDG) control and instrumentation is discussed in Subsection 8.3.1.1.8 of the PSAR and will be covered by a review of Chapter 8.

The emergency diesel generator support systems are described in subsections of Chapter 9 of the PSAR and consist of the following:

1) Emergency diesel generator jacket water system
2) Emergency diesel generator starting air system
3) Emergency diesel generator lubrication system
4) Emergency diesel fuel storage and transfer system
5) Emergency diesel combustion air intake and exhaust system

The instrumentation and controls for the emergency diesel generator and its auxiliaries and support systems are designed to provide the automatic startup and continued operation of the emergency diesel generator units of the plant standby power system under emergency or DBA conditions.

The specific regulatory requirements applicable to the emergency diesel generator and its auxiliaries are listed in Table 7.1-1 (Table 7.1-2 of the PSAR).

7.3.1.7 Reactor Building Cooling Water System (RBCW) and Reactor Building Service Water System (RBSW) Instrumentation and Controls

The control systems for the Reactor Building Cooling Water (RBCW) System and Reactor Building Service Water (RBSW) System operate to maintain the flow of cooling water to operate auxiliaries which are required for normal plant operation and normal or emergency reactor shutdown, as well as to those auxiliaries whose operation is desired following a LOCA but are not essential to safe shutdown. The RBCW System is comprised of three divisions. The RBSW System is also comprised of three divisions. The power for RBCW and RBSW instrumentation and controls is supplied from Division I, II, and III 125 VDC and 120 VAC essential power buses.

During normal operation, RBCW water flows through safety-related and non-safety-related equipment except the RHR and emergency diesel generator heat exchangers. During all plant operating modes, one RBCW pump and one RBSW pump are normally operating in each division, so that in the event of LOCA, the RBCW and RBSW required to shut down the plant safely are already in operation. To protect against flooding of the basement of the Control Building, the portions of the RBSW external to the Control Building are isolated by automatic valves.

Isolation of the non-safety-related section of each division of the RBCW System from the safety-related section is accomplished by motor operated valves in the inlet and outlet lines to the non-safety-related section. Flow sensors are located in the inlet lines to the non-safety related section.

The instrumentation and controls of these systems are designed to the following functional requirements:

1. Maintain control of cooling water to equipment that requires cooling during reactor shutdown modes and following a LOCA or LOOP or both.

2. Provide for the automatic isolation of the non-essential parts of the Reactor Building Cooling Water (RBCW) System from the essential parts during a LOCA or upon detection of a major RBCW leak in the non-essential portion.
3. Provide for the automatic isolation of all parts of the Reactor Building Service Water (RBSW) Division outside of the Control Building upon receipt of a high water level signal in the RBCW Heat Exchanger Room in that division.

Specific regulatory requirements applicable to the system instrumentation and controls are given in Table 7.1-1 (Table 7.1-2 of the PSAR).

7.3.1.8 Essential HVAC Systems Instrumentation and Controls

Subsections 9.4.1 (Control Building HVAC), 9.4.5 (Reactor Building HVAC), and 9.4.15 (Auxiliary Fuel Building HVAC) of the PSAR describe the I&C for these systems. A review of Chapter 9 of the PSAR will cover these.

7.3.1.9 HVAC Emergency Chilled Water System Instrumentation and Control

The Emergency Chilled Water System (ECW) supplies demineralized chilled water to the cooling coils of the Control Building safety-related electrical equipment (CBSREE) rooms and control room habitability area (CRHA) air handling units, and the Reactor Building safety-related equipment area (RBSREA) air conditioning systems. The system resides in three divisions, each containing two chillers and two chilled water pumps. The instrumentation and controls of ECW are supplied with 120 VAC and 125 VDC electric power from Division I, II, and III power buses.

ECW consists of three mechanically (and electrically) separate subsystems -- Divisions A, B, and C. The system is designed to provide chilled water to the cooling coils of CBSREE, CHRA, and RBSREA. ECW is designed to operate during both accident conditions and normal plant operation, and during all modes of operation for the cooling systems it serves. Each division of ECW consists of two chilled water pumps and two chillers; each chiller includes the condenser, evaporator, compressor, refrigerant piping and package chiller controls. The system condenser is cooled by the Reactor Building Cooling Water System (RBCW). ECW shares a surge tank with the RBCW. Lack of flow of RBCW to the refrigerant condenser automatically stops the chiller. A flow transmitter, provided at the chilled water line, shuts down the chiller and chilled water pump upon indication of low flow in the chilled water line.

The Emergency Chilled Water System (ECW) instrumentation and controls are designed such that it provides controls for cooling units that ensure a controlled environment for essential equipment and control room areas following a loss-of-

coolant accident, loss of offsite power, or isolation of normal heating, venting, and air conditioning (HVAC). During detailed design, a site-specific MCR temperature rise analysis is to be performed for the Station Blackout (SBO) scenario.

Specific regulatory requirements applicable to the system instrumentation and control are given in Table 7.1-1 (Table 7.1-2 of the PSAR).

7.3.1.10 Nitrogen Supply System (N2) Instrumentation and Controls

The Nitrogen Supply System (N2) provides compressed nitrogen of the required pressure to the ADS SRVs, the MSIVs (for testing only), instruments and pneumatically operated valves in the Primary Containment Vessel (PCV) and other nitrogen-using components in the reactor building. The safety-related portion of the N2 is powered from the onsite Class 1E AC and DC systems. N2 Division A is powered from Class 1E Division I, and N2 Division B is powered from Class 1E Division II. The safety-related portion is switched automatically to the standby vital AC power supply during a loss of normal power. The non-safety-related portion is connected to the normal AC power supply.

The N2 is separated into non-safety -related and safety-related sections. The non-safety-related portion of the system includes an inlet filter, piping, and valves to all nitrogen users. The safety-related portion of the system includes two banks of high pressure nitrogen bottles and associated piping, valves, and controls. When low nitrogen gas pressure is detected in the lines to the ADS accumulators the safety-related portion of the system is isolated from the non-safety-related portion by isolation valves which automatically cut off the normal nitrogen gas supply and open the emergency nitrogen gas bottle supply to the ADS accumulators.

In addition to valves that isolate non-safety-related equipment from safety-related equipment, the N2 is provided with containment isolation valves where the N2 lines enter containment. The valves are manually operated from individual control switches in the MCR.

The instrumentation and controls of the N_2 are designed such that it provides automatic and manual control of the nitrogen gas supply to assure its operation during all modes of plant operation, and to automatically initiate the emergency nitrogen bottle supply (on low nitrogen supply pressure) to assure adequate supply of nitrogen to automatic depressurization safety/relief valves (SRVs) and to nitrogen-using equipment and valves in the reactor building.

Specific regulatory requirements applicable to this system are listed in Table 7.1-1 (Table 7.1-2 of the PSAR).

7.3.1.11 Flammability Control System Instrumentation and Controls

The FCS provides control over hydrogen and oxygen generated following a LOCA. In an inerted containment, mixing of any hydrogen generated is not required. Any oxygen evolution from radiolysis is very slow such that natural convection and molecular diffusion are sufficient to provide mixing. Spray operation will provide further assurance that the drywell or wetwell is uniformly mixed. FCS consists of the following features:

1) FCS has two recombiners installed in the secondary containment. The recombiners process the combustible gases drawn from the primary containment drywell.

2) FCS is activated when a LOCA occurs. The oxygen and hydrogen remaining in the recombiners after having been processed are transmitted to the suppression pool.

The safety-related components of FCS inside the Reactor Building are protected from postulated missiles and pipe whip, as required to assure proper functional integrity. FCS is designed to withstand the dynamic effects associated with a safe shutdown earthquake (SSE) without loss of isolation function. FCS is designed so that all components exposed to the PCV atmosphere are capable of performing their safety-related function under the temperature, humidity, pressure, and radiation transients associated with a design basis LOCA. The FCS equipment valves and instrumentation and controls are designed to withstand the thermodynamic and radiological environmental conditions inside the Reactor Building secondary containment during a design basis LOCA. FCS is capable of being functional tested to verify the ability of the system to perform its safety function.

Specific regulatory requirements applicable to this system are listed in Table 7.1-1 (Table 7.1-2 of the PSAR).

7.3.2 Evaluation Findings

The review of the instrumentation and control aspects of the engineered safety feature (ESF) systems included the engineered safety features actuation systems (ESFAS) and the ESF control systems. The ESFAS detects a plant condition requiring the operation of an ESF system and/or essential auxiliary supporting (EAS) system and initiates operation of these systems. The ESF control systems regulate the operation of the ESF systems following automatic initiation by the protection system or manual initiation by the plant operator.

The review concludes that the design of the ESFAS is acceptable and meets the relevant requirements of General Design Criteria (GDC) 1, 2, 4, 13, 19-24 and 29 and 10CFR50.34(f), 10CFR50.55a(a)(1), and 10CFR50.55a(h).

The review of these systems was for conformance to the guidelines in the regulatory guides, industry standards and branch technical positions applicable to these systems. The review concludes that the PSAR acceptably identified the guidelines applicable to these systems. Based upon the review of the system design for conformance to the guidelines, the review concludes that the systems conform to the guidelines applicable to these systems. Therefore the requirements of GDC 1 and 10CFR50.55a(a)(1) have been met.

The review included the identification of those systems and components of the ESFAS and ESF control systems that are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. The review concludes that these systems and components have been identified consistent with the design bases. The qualification programs to demonstrate the capability of these systems and components to survive the above effects are addressed in other Sections of the PSAR and are assumed acceptable. Therefore the review finds that the identification of these systems and components satisfies the requirements of GDC 2 and 4.

Based on the review of ESFAS and ESF control system status information, manual initiation capabilities, control capabilities, and provisions to support safe shutdown, the review concludes that information is provided to monitor the system over the anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety. Appropriate controls are provided for manual initiation and control of ESF functions. ESF controls appropriately support actions to operate the nuclear power unit safely under normal conditions, and to achieve and maintain a safe condition under accident conditions. Therefore, the review finds that the ESFAS and ESF control design satisfies the requirements of GDC 13 and 19.

The review of system functions concludes that the ESFAS conforms to the requirements of ANSI/IEEE Std 279 and 10CFR50.43(f). The ESFAS setpoint methodology conforms to the guidance of Draft Reg. Guide DG-1045. This review concludes that the ESFAS includes the provision to sense accident conditions and anticipated operational occurrences consistent with the accident analysis presented in Chapter 15 of the SAR. Therefore, the review finds that the ESFAS satisfies the requirements of GDC 20.

The ESFAS conforms to the guidelines for periodic testing in Reg. Guide 1.22 and Reg. Guide 1.118. The bypassed and inoperable status indication conforms to the guidelines of Reg. Guide 1.47. The ESFAS conforms to the guidelines on the

application of the single-failure criterion in ANSI/IEEE Std 379 as supplemented by Reg. Guide 1.53. The review concludes that the ESFAS satisfies the requirement of ANSI/IEEE Std 279 with regard to system reliability and testability. Therefore, the review finds that the ESFAS satisfies the requirements of GDC 21.

The ESFAS conforms to the guidelines in Reg. Guide 1.75 for protection system independence. The review concludes that the ESFAS satisfies the requirement of ANSI/IEEE Std 279 with regard to the system's independence. Therefore, the review finds that the ESFAS satisfies the requirements of GDC 22.

The review of the ESFAS concludes that the system is designed to fail into a safe state if conditions such as disconnection of the system, loss of energy, or postulated adverse environments are experienced. Therefore, the review finds that the ESFAS satisfies the requirements of GDC 23.

The review of the interfaces between the ESFAS and plant control systems concludes that the system satisfies the requirements of ANSI/IEEE Std 279 with regard to control and protection system interactions. Therefore, the review finds the ESFAS satisfies the requirements of GDC 24.

The ESF control systems were reviewed for conformance to the requirements for testability, operability with onsite and offsite electrical power, and single failures. The review concludes that the ESF control systems are testable and are operable using either onsite or offsite power (assuming only one source is available).

The conclusions noted above are based upon the requirements of ANSI/IEEE Std 279 with respect to the design of the ESFAS. Therefore, the review finds that the ESFAS satisfies the requirements of 10CFR50.55a(h).
The review of the ESFAS examined the dependence of this system on the availability of essential auxiliary systems. The review concludes that the design of the ESFAS is compatible with the functional requirements of EAS systems.

The review of software development plans concludes that the computer systems meet the guidance of Reg. Guide 1.152. The special characteristics of computer systems have been adequately addressed and the review finds that the ESFAS satisfies the requirements of GDC 1 and 21.

The review of the PSAR commitments to defense-in-depth and diversity concludes that the ESFAS complies with the criteria for defense against common-mode failure in digital instrumentation and control systems. Therefore the review finds that adequate diversity and defense against common-mode failure has been provided to satisfy these requirements of GDC 21, GDC 22, and the USNRC Requirements Memorandum on SECY-93-087.

7.4 Safe Shutdown Systems (PSAR Section 7.4)

7.4.1 Introduction

The objectives of the safe shutdown systems review were to confirm that the safe shutdown systems satisfy the requirements of the acceptance criteria and guidelines applicable to safety systems, and that they are designed to perform their safety functions during all plant conditions for which they are required. Both kinds of shutdown conditions, hot shutdown and cold shutdown, were addressed. The acceptance criteria and guidelines applicable to safe shutdown systems identified in SRP Section 7.4 were used in the review. The review confirmed that these systems conform to the requirements of the acceptance criteria and guidelines.

7.4.2 Safe Shutdown Systems Description

PSAR section 7.4 discusses the instrumentation and control aspects of the following plant systems and functions designed to assure safe and orderly shutdown of the reactor:

1) Reactor Shutdown Cooling (SDC) mode of RHR
The SDC mode of the RHR System is used during the normal or emergency reactor shutdown and cooldown. The initial phase of the SDC mode is accomplished following insertion of the control rods and steam blowdown to the main condenser which serves as the heat sink. Reactor shutdown cooling has three independent loops. Each loop consists of pumps, valves, a heat exchanger, and instrumentation designed to provide decay heat removal capability for the core. This mode specifically accomplishes the following:

   a. Reactor Shutdown - in conjunction with other systems removes enough residual heat (decay and sensible) from the reactor vessel water to cool it to 60°C with all three SDC loops available within 24 hours after the control rods are inserted. This mode is manually activated with the reactor pressure below 0.93 MpaG.
   b. Safe Shutdown (Emergency shutdown) brings the reactor to a cold shutdown condition (<100°C) with two-out-of-three shutdown cooling loops available within 36 hours after control rod insertion. This mode is manually activated with the reactor pressure below 0.93 MpaG.

   The RHR mode can accomplish its design objective by a preferred means of directly extracting reactor vessel water from the vessel shutdown nozzle and routing it to a heat exchanger and back to the vessel. After extracting heat via the RHR heat exchanger this cooler water is returned to the vessel via the feedwater line (Loop A) and via the core cooling injection nozzles (Loops B and C).

Electrical components for the reactor SDC mode of the RHR System are safety-related and are classified as Class 1E. This system utilizes plant Class 1E power sources. These include 4.16 KV for the pumps, and 480 VAC for other motors (both backed by the DG), and 120 VAC instrument buses (backed by DC sources).

The reactor water is cooled by taking suction from the three SDC suction nozzles. Water is pumped through the respective RHR system heat exchanger and back to the reactor vessel via the feedwater lines (Loop A) and the LPFL injection nozzles (Loops B and C). If it is necessary to discharge a complete core load of reactor fuel to the fuel pool, a means is provided for making a physical inter-tie between the Spent Fuel Pool Cooling and Cleanup (FPCU) System and the RHR heat exchangers. This increases the cooling capacity of FPCU to handle the heat load for this situation. The fuel pool inter-tie is applied only to Loops B and C. The reactor Shutdown Cooling System is initiated by manual operator actions.

The shutdown cooling mode of the RHR monitoring and controls provide the following capabilities:

a. Enable the system to remove the residual heat (decay heat and sensible heat) from the reactor vessel during emergency shutdown when the vessel pressure is below approximately 931 kPaG.
b. Provide manual controls for the shutdown cooling system in the MCR and at the RSD panel.
c. Indicate performance of the shutdown cooling system by MCR instrumentation and controls in the RSD panel.

Specific regulatory requirements applicable to reactor shutdown cooling are given in Table 7.1-1 (Table 7.1-2 of the PSAR).

2) Remote Shutdown System

The Remote Shutdown System (RSD) provides a means to carry out the reactor shutdown functions from outside the main control room (MCR) and bring the reactor to hot shutdown and subsequent cold shutdown through suitable procedures, in a safe and orderly fashion.

The RSD provides the capability for remote control of reactor systems needed to carry out the shutdown function from outside the MCR and bring the reactor to hot shutdown and subsequent cold shutdown through suitable procedures. The RSD also provides a variation to the normal system used in the MCR permitting the shutdown of the reactor when feedwater is unavailable and the normal heat sinks (turbine and condenser) are lost.

Reactor pressure will be controlled and core decay and sensible heat rejected to the suppression pool by relieving steam pressure through the automatic

activation of relief valves. Reactor water inventory will be maintained by the HPCF System. During this phase of shutdown, the suppression pool will be cooled by operating the RHR System in the Suppression Pool cooling mode.

Manual operation of the relief valves will cool the reactor and reduce its pressure at a controlled rate until reactor pressure becomes so low that HPCF System operation is discontinued. The RHR System will then be operated in the Shutdown Cooling mode using the RHR System heat exchanger in the reactor water circuit to bring the reactor to the cold low pressure condition.

The Remote Shutdown System (RSD) instrumentation and controls provide the following:

a. Instrumentation and controls outside the MCR to allow prompt hot shutdown of the reactor after a scram and to maintain safe conditions during hot shutdown.
b. Capability for subsequent cold shutdown of the reactor through the use of suitable procedures.
c. Protective functions in the event the MCR becomes uninhabitable.

Specific regulatory requirements applicable to the remote shutdown system are listed in Table 7.1-1 (Table 7.1-2 of PSAR).

3) Standby Liquid Control System (SLC)

The instrumentation and controls for SLC are designed to initiate and continue injection of a liquid neutron absorber (boron solution) into the reactor when either manually or automatically called upon to do so. This equipment also provides the necessary controls to maintain this liquid chemical solution well above saturation temperature in readiness for injection.

The SLC provides a backup method to shut down the reactor to cold subcritical conditions by independent means other than the normal method by the CRD System. Thus, the system is considered a safe shutdown system. The SLC process equipment, instrumentation and control essential for injection of the neutron absorber solution into the reactor is classified as nuclear safety related. The SLC I&C and equipment are powered from 1E sources.

The instrumentation and controls of the Standby Liquid Control System (SLC) equipment provide for shutting the reactor down from full power to cold shutdown and maintaining the reactor in a subcritical state at atmospheric temperature and pressure conditions by pumping sodium pentaborate (a neutron absorber) into the reactor.

Specific regulatory requirements applicable to this system are given in Table 7.1-1 (Table 7.1-2 of PSAR).

7.4.3 Evaluation Findings

The review concludes that the design of the safe shutdown systems and the safe shutdown initiation of the essential auxiliary support (EAS) systems are acceptable and meet the relevant requirements of General Design Criteria (GDC) 1, 2, 4, 13 and 19, and 10CFR50.55a(a)(1).

The review of these systems was conducted for conformance to the guidelines in the regulatory guides and standards applicable to these systems. The review concludes that the PSAR adequately identified the guidelines applicable to these systems. The review of the system design for conformance to the guidelines finds that there is reasonable assurance that the systems fully conform to the guidelines applicable to these systems. Therefore, the review finds that the requirements of GDC 1 and 10CFR50.55a(a)(1) have been met.

The non-safety portions of information systems important to safety are appropriately isolated from safety systems, including the safety portions of the information systems. Therefore, the review concludes that the isolation of these systems from safety systems satisfies the requirements of 10CFR50.55a(h) and the requirements of GDC 24.

The review included the identification of those systems and components for the safe shutdown systems which are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments and missiles. Based upon the review, the review concludes that the PSAR has identified those systems and components consistent with the design bases for those systems. The qualification programs to demonstrate the capability of these systems and components to survive these events are addressed in other sections of the PSAR and are assumed adequate. Therefore, the review finds that the identification of these systems and components satisfies the requirements of GDC 2 and 4.

The review concludes that instrumentation and controls have been provided to maintain variables and systems which can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems within prescribed operating ranges during plant shutdown. Therefore, the review finds that the systems required for safe shutdown satisfy the requirements of GDC 13.

Instrumentation and controls have been provided within the control room to allow actions to be taken to maintain the nuclear power unit in a safe condition during shutdown, including a shutdown following an accident. Equipment at appropriate locations outside the control room has been provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during the hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures. Therefore, the

review concludes that systems required for safe shutdown satisfy the requirements of GDC 19.

The review of the instrumentation and control systems required for safe shutdown includes conformance to the requirements for testability, operability with onsite and offsite electrical power, and single failures as appropriate based on their safety function consistent with the General Design Criteria applicable to safe shutdown systems. The review concludes that these systems are designed to be testable and operable on either onsite or offsite electrical power, and that the controls associated with redundant safe shutdown systems are independent and satisfy the requirements of the single-failure criterion and, therefore, meet the relevant requirements of GDC 34, 35, and 38.

The review of the safe shutdown systems examined the dependence of these systems on the available essential auxiliary systems. This review concludes that the design of the safe shutdown systems is compatible with the functional requirements of EAS systems. The review of software development plans concludes that the computer systems design process meets the guidance of Reg. Guide 1.152. Therefore, the special characteristics of computer systems have been adequately addressed, and the review finds that the computer-based safe shutdown systems satisfy the requirements of GDC 1.

7.5 Information Systems Important to Safety (PSAR Section 7.5)

7.5.1 Introduction

The objectives of the review of information systems important to safety were to confirm that the information systems important to safety satisfy the requirements of the acceptance criteria and guidelines applicable to these systems, and that they are designed to provide the information to ensure plant safety during all plant conditions for which they are required. These systems include:

- Post-accident monitoring (PAM) systems
- Process radiation monitoring system
- Containment monitoring system

The acceptance criteria and guidelines applicable to information systems important to safety identified in SRP Section 7.5 were used in the review.

7.5.2 Information Systems Important to Safety Descriptions

Section 7.5 of the PSAR describes the information systems important to safety. Safety-related display systems are those systems which provide information for the safe operation of the plant during normal operation, anticipated operational occurrences, and accidents. The information systems important to safety include those systems which provide information for manual initiation and control of safety systems, to indicate that

plant safety functions are being accomplished and to provide information from which appropriate actions can be taken to mitigate the consequences of anticipated operational occurrences and accidents. The Safety Parameter Display System (SPDS), information systems associated with the emergency response facilities and nuclear data link are information systems important to safety.

The annunciator system for Lungmen NPS meets applicable requirements for redundancy, independence, and separation. In addition, alarms that are provided for manually controlled actions for which no automatic control is provided and that are required for the safety systems to accomplish their safety functions meet the applicable requirements for Class 1E equipment and circuits.

7.5.2.1 Post Accident Monitoring System (PAM)

Regulatory Guide 1.97 defines five "types" and three "categories" of plant variables for accident monitoring instrumentation. Each variable has been defined as to both type and category. Plant variables are divided into types according to the purpose of the indication to the plant operator. Table 7.5-2 of the PSAR provides a list of the Lungmen PAM variables.

The detailed Design and Qualification Criteria for Category 1, 2, and 3 variables are provided in Reg. Guide 1.97 for:

1. Equipment Qualification
2. Redundancy
3. Power Sources
4. Channel Availability
5. Quality Assurance
6. Display and recording
7. Range
8. Equipment Identification
9. Interfaces
10. Servicing, Testing and Calibration
11. Human Factors
12. Direct Measurement

A detailed listing of the design and qualification criteria for Categories 1, 2, and 3 is provided in Table 7.5-1 of the PSAR.

In addition to design and qualification criteria, Regulatory Guide 1.97 provides a comprehensive listing of "BWR variables" which address accident monitoring requirements. Table 7.5-2 of the PSAR was developed using Table 2 of Regulatory Guide 1.97 as a guide. Design and qualification criteria are addressed as category designations per the discussion above. Variables listed in Table 7.5-2 without comment

meet the design and qualification requirements of Regulatory Guide 1.97. Any exceptions taken are noted in the comment column.

The Post Accident Monitoring (PAM) system is designed to provide the necessary display instrumentation in the MCR so the reactor operator can determine and accomplish the manual control actions required for safe plant operation during post accident events.

7.5.2.2 Process Radiation Monitoring System (PRM)

A number of radiation monitoring functions are provided on process lines, HVAC ducts, and vents that may serve as discharge or intake routes for radioactive materials. These include the following:

1) Main steam line tunnel
2) Reactor Building heating, ventilating, and air conditioning (HVAC) exhaust
3) Fuel handling area HVAC exhaust
4) Control Building HVAC air intake supply
5) Drywell sumps discharge
6) Offgas pre-treatment
7) Offgas post-treatment
8) Charcoal vault ventilation exhaust
9) Ventilation stack discharge
10) Radwaste liquid discharge
11) Reactor Building cooling water
12) Radwaste Building ventilation exhaust
13) Turbine Building ventilation exhaust
14) Turbine gland seal condenser exhaust
15) Standby gas treatment system exhaust
16) Drywell fission product
17) Access Control Building ventilation intake
18) Plant stack discharge
19) Containment Overpressure Protection System discharge

A simplified diagram depicting the process radiation subsystems is shown in Figure 11.5-1 of the PSAR. Subsystems (1) through (5) are classified nuclear safety-related, while subsystems (6) through (19) are classified as non-safety-related. System descriptions and requirements are described in detail in Section 11.5 of the PSAR.

The PRM is designed to:

1. Monitor the gross radiation level in the main steamlines tunnel area and in the ventilation discharge ducting of the primary and secondary containment structures.

2. Provide radiation measurement, display, recording and alarm capability in the MCR.
3. Provide alarm annunciation signals to the main control room if alarm or trip levels are reached or the subsystem is in an inoperative condition.
4. Provide channel trip inputs to the Safety System Logic and Control (SSLC) to initiate shutdown of the mechanical vacuum pump and closure of the mechanical pump discharge line isolation valve.
5. Provide trip signals to isolate the secondary containment, and to initiate the SGT on high radiation levels in the exhaust ducts of the fuel handling area or in the Reactor Building.
6. Monitor the intake air supply to the Control Building so habitability of the MCR can be maintained during an accident condition.

7.5.2.3 Containment Monitoring System Instrumentation and Controls

The Containment Monitoring System (CMS) consists of the following functions:

1) Containment Atmospheric Monitoring (CAM)
2) Post Accident Sampling (PAS)
3) Suppression Pool Temperature Monitoring (SPTM)
4) Drywell pressure and temperature measurement
5) Drywell and wetwell dew points measurement
6) Primary containment Integrated Leakage Rate Test (ILRT) instrumentation
7) Suppression pool water level measurement

CMS includes components which monitor processes inside the Primary Containment Vessel (PCV), inside the reactor building and outside the secondary containment.

The CAM portion of CMS consists of two independent and redundant Class 1E electrical divisions that monitor the gamma radiation dose rate and the concentration of hydrogen and oxygen in the drywell and wetwell air during plant operation and following a Loss-of-Coolant Accident (LOCA) event.

Each radiation monitoring channel consists of a gamma sensitive Radiation Detection Assembly and a Signal Conditioning Unit. The Radiation Detection Assemblies are located at widely separated locations within the PCV volume to provide representative sampling of the PCV volume. The radiation levels are transmitted to the Plant Computer System (PCS). This information will be communicated to the Main Control Room (MCR) personnel as determined by a Human Factors evaluation.

The PAS portion of CMS, located in the secondary containment, allows sampling of the reactor water and PCV air during post-accident conditions.

The SPTM portion of CMS measures the average bulk suppression pool temperature. The temperature elements are Class 1E and arranged circumferentially around the

suppression pool. The SPTM portion of CMS consists of four redundant electrical divisions with four levels of temperature elements within each electrical division.

The drywell pressure and temperature instrumentation consists of pressure transmitters and temperature elements located throughout the PCV. The drywell is provided with seven pressure transmitters while the wetwell has three transmitters. Four of the drywell and wetwell volumes are also provided with two differential pressure transmitters, connected between these two volumes. The piping for these instruments also contain two transmitters used to measure drywell flooding in the event of severe accidents.

Forty-eight non-safety related temperature elements, located throughout the upper and lower drywell, are provided for measuring the drywell temperature. In addition, there are four additional temperature elements, spaced at approximately equal intervals in the vertical direction of the drywell to monitor ambient temperatures over the drywell's full height. These additional temperature elements provide indication and alarm for Leak Detection and Isolation System use.

Thirteen non-safety-related dew point elements, located throughout the upper and lower drywell, are provided for measuring the drywell dew points. These dew point elements, in conjunction with the non-safety-related temperature elements, are used for PCV absolute pressure calculations during containment Integrated Leakage Rate Test (ILRT).

Suppression pool water level monitoring is provided to measure the inventory of suppression pool water. Suppression pool water level monitoring consists of six channels of water level detection sensors distributed into narrow and wide range instruments. The narrow range suppression pool water level signal is used to detect the uncovering of the first set of suppression pool temperature sensors below the pool surface. When the suppression pool water level drops below the elevation of a particular temperature sensor, that particular temperature sensor signal is not used in computing the average pool temperature.

The Containment Monitoring System (CMS) is designed to:

1. Monitor hydrogen, oxygen concentration, and gamma radiation levels in the drywell and wetwell air during normal plant operation and post accident conditions. Provide annunciation if radiation, hydrogen and oxygen alarm levels are reached or if a radiation, hydrogen and oxygen monitoring channel is in an inoperative state.
2. Monitor suppression pool (S/P) temperatures and upon exceeding a high limit, initiate suppression pool cooling mode of RHR and provide signals to the RPS for scram use. The suppression pool temperature monitoring (SPTM) function also isolates RBCW to the Reactor Water Cleanup System (RWCU) non-regenerative heat exchanger for heat load shedding to increase suppression pool cooling.
3. Provide drywell pressure signals for LOCA initiation, and for use by RPS for reactor scram, and for use by LDI for primary containment vessel (PCV) isolation.

4. Lower drywell flooding (LDF) functions to flood the lower drywell with water from the suppression pool in the unlikely event of a severe accident where the core is postulated to melt and cause a subsequent vessel failure.

The specific regulatory requirements applicable to the information systems are listed in Table 7.1-1 (Table 7.1-2 of the PSAR).

7.5.3 Evaluation Findings

The review concludes that the designs of the information systems important to safety are acceptable and meet the relevant requirements of General Design Criteria(GDC) 1, 2, 4, 13, 19, and 24, and 10CFR50.34(f), 10CFR50.55a(a)(1), and 10CFR50.55a(h).

The review of the information systems important to safety was conducted for conformance to the guidelines in the regulatory guides and industry codes and standards applicable to these systems. The review concluded that the PSAR adequately classified and identified the guidelines applicable to these systems. The review of the system design for conformance to the guidelines finds that the systems conform to the guidelines applicable to these systems. Therefore, the review finds that the requirements of GDC 1 and 10CFR50.55a(a)(1) have been met.

The review included the identification of those systems and components for the information systems important to safety that are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. The review concludes that the PSAR has identified those systems and components consistent with the design bases for those systems. Qualification programs to demonstrate the capability of these systems and components to survive these events are covered in other parts of the PSAR and are assumed adequate. Therefore, the review finds that the identification of these systems and components satisfies the requirements of GDC 2 and 4.

The non-safety portions of information systems important to safety are appropriately isolated from safety systems, including the safety portions of the information systems. Therefore the review concludes the isolation of these systems from safety systems satisfies the requirements of 10CFR50.55a(h) and the requirements of GDC 24.

The instrumentation provided for monitoring severe accident conditions has been designed to operate in the severe-accident environment for which they are intended and over the time span for which they are needed. Therefore, the review finds that the severe accident monitoring instrumentation satisfies the requirements of GDC 2 and 4.

The post-accident monitoring system conforms to the guidelines for the instrumentation to access plant conditions during and following an accident provided in Reg. Guide 1.97. The redundant information systems conform to the guidelines for the physical independence of electrical systems provided in Reg. Guide 1.75. The instrument spans and EOP actions points are to be established in accordance with the guidelines of Reg. Guide 1.105. The environmental monitoring system provided to protect the safety instrument sensing lines from freezing conforms to the guidelines of Reg. Guide 1.151. The post-accident monitoring system includes appropriate variables. The range and accuracy of the instrument channels for these variables are consistent with the plant safety analysis. The post-accident monitoring system includes appropriate variables for monitoring severe accident conditions. The variables monitored and the range and accuracy of instrumentation provided to monitor these variables is consistent with the severe accident analysis. Therefore the review finds that the post-accident monitoring system meets the requirements of GDC 13 and 19.

The post-accident monitoring system includes the following functions required by 10CFR50.34(f): accident monitoring instrumentation, inadequate core cooling instrumentation, instruments for monitoring plant conditions following core damage, central reactor vessel water level recording. Therefore, the review concludes that the instrumentation systems important to safety satisfy the applicable requirements of 10CFR50.34(f).

The bypass indications satisfy the guidelines of Reg. Guide 1.47. Therefore, the review concludes that the bypassed or inoperable status indication functions satisfy the applicable requirements of 10CFR50.55a(h) and 10CFR50.34(f)(2)(v).

The control room annunciation was reviewed and found adequately designed to support normal and emergency plant operations. Therefore, the review concludes that the annunciation satisfies the requirements of GDC 13 and 19.

In summary, the review finds that the information systems satisfy the requirements of GDC 13 for monitoring variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions. Further, the review finds that conformance to GDC 13 and the applicable guidelines satisfies the requirements of GDC 19 with respect to information systems provided in the control room from which actions can be taken to operate the unit safely under normal conditions and to maintain it in a safe condition under accident conditions.

The safety parameter display system, the information systems associated with the emergency response facilities, and the nuclear data link, non-safety portions of PAM, non-safety portions of BISI, and non-safety portions of the annunciator systems are appropriately isolated from safety systems. Electrical isolation devices

are to be qualified in accordance with the guidance of BTP HICB-11. Therefore, the review concludes that the isolation of these systems from safety systems satisfies the requirements of 10CFR50.55a(h) and GDC 24.
The design also incorporates the recommendations of TMI task action plan items and are acceptable.

Based on the review of software design and development plans, the review concludes that the computer systems meet the guidance of Reg. Guide 1.152. Therefore, the special characteristics of computer systems have been adequately addressed, and the review finds that the information systems important to safety satisfy the requirements of GDC 1.

7.6 Interlock Systems Important to Safety (PSAR Section 7.6)

7.6.1 Introduction

The objective of the review of interlock systems important to safety was to confirm that design considerations such as redundancy, independence, single failures, qualification, bypasses, status indication, and testing are consistent with the design bases of these systems and commensurate with the importance of the safety functions to be performed.

The review encompassed those interlock systems important to safety that operate to reduce the probability of occurrence of specific events, or to maintain safety systems in a state that ensures their availability in an accident. These systems include interlock systems to prevent over pressurization of low-pressure systems, for example, residual heat removal (RHR) when these systems are connected to high-pressure systems (for example, primary coolant), interlocks to prevent over pressure of the primary coolant system during low-temperature operation of the reactor vessel, valve interlocks to ensure the availability of emergency core cooling system (ECCS) accumulators, interlocks to isolate safety systems from non-safety systems (for example, seismic and non-seismic portions of auxiliary supporting systems), and interlocks to preclude inadvertent inter-ties between redundant or diverse safety systems where such inter-ties exist for the purposes of testing or maintenance.

The acceptance criteria and guidelines applicable to interlock systems identified in SRP Section 7.6 were applied.

7.6.2 Description of Interlock Systems Important to Safety

Section 7.6 of the PSAR discusses the instrumentation and control aspects of the following plant systems:

1) RHR High Pressure/Low Pressure (HP/LP) Interlocks

The low pressure modes of the RHR System which connect to the reactor coolant pressure boundary (RCPB) and the instrumentation which protects them from overpressurization are discussed in this section. Such high pressure/low pressure (HP/LP) interfaces with the reactor vessel are exclusive to the RHR System. The power for the interlocks is provided from the essential power supplies used for the RHR System and its various modes of operation. At least two valves are provided in series in each of these lines. The RHR shutdown cooling supply valves have independent sets of interlocks to prevent the valves from being opened when the primary system pressure is above the subsystem design pressure or when reactor water level is below Level 3. These valves also receive a signal to close when reactor pressure is above system pressure, or reactor water level is below Level 3. An additional interlock is RHR equipment area ambient temperature.

The RHR shutdown cooling/LPRL injection valve is interlocked to prevent valve opening whenever the reactor pressure is above the subsystem design pressure, and automatically closes whenever the reactor pressure exceeds the subsystem design pressure. This valve must operate for long-term cooling, and has a remote testable check valve downstream. The check valve operation can be confirmed provided the RHR LPFL injection valve is closed.

The High Pressure/Low Pressure Interlock is designed to protect the low pressure system boundary from postulated overpressurization from the reactor system.

The wetwell-to-drywell vacuum breaker interlocks are described in Subsection 6.2.1 of the PSAR and will be covered by a Chapter 6 review.

2) Atmospheric Control System (ACS) Vacuum Breaker Interlocks
The atmospheric control systems vacuum breaker interlocks are described in Subsection 6.2.1 of the PSAR and will be covered by a Chapter 6 review.

7.6.3 Evaluation Findings

The review concludes that the design of the interlock systems is established in accordance with its safety function, is acceptable, and meets the relevant requirements of General Design Criteria (GDC) 1, 2, 4, 10, 13, 15, 16, 19 and 24, 10CFR50.55a(a)(1) and 10CFR50.55a(h).

The review of these systems was made for conformance to the guidelines in the regulatory guides and industry codes and standards applicable to these systems. The review concluded that the PSAR adequately identified the guidelines applicable to these systems and has properly classified them. Based upon the review of the system design for conformance to the guidelines, the review finds that there is reasonable assurance that the systems fully conform to the guidelines applicable to these systems. Therefore the

review finds that the requirements of GDC 1, 15 and 16, and 10CFR50.55a(a)(1) have been met.

The review of interlock system functions concluded that appropriate interlocks are provided to maintain an appropriate design margin to assure that acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Therefore, the review finds that the interlock systems satisfy the requirements of GDC 10, 15 and 16.

The review of interlock system status information, initiation capabilities, and provisions to support safe shutdown concluded that information is provided to monitor interlocks over the anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety. Appropriate controls are provided for interlock initiation and bypass. The interlocks appropriately support actions to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions. Therefore, the review finds that the interlock systems satisfy the requirements of GDC 13 and 19.

The review of these systems found that they comply with the reliability guidance of IEEE Std 279.

The review included the identification of those systems and components for the interlock systems that are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. The review concludes that the PSAR has identified those systems and components consistent with the design bases for those systems. The qualification programs to demonstrate the capability of these systems and components to survive these events are covered elsewhere in the PSAR and are assumed will be covered by reviews of the other Chapters of the PSAR. Therefore, the review finds that the identification of these systems and components satisfies the requirements of GDC 2 and 4.

The bypassed or inoperable status indication of safety interlocks were reviewed. The review found that appropriate bypass indications are provided to give the operators timely information and status reports so the operators can mitigate the effects of unexpected system unavailability. The bypass indications satisfy the guidelines of Reg. Guide 1.47. Therefore, the review concludes that the safety interlock systems satisfy the applicable requirements of 10CFR50.55a(h) and 10CFR50.34(f)(2)(v).

The non-safety interlock systems are appropriately isolated from safety systems. Therefore, the review concludes that the isolation of these systems from safety systems satisfies the requirements of 10CFR50.55a(h) and the requirements of GDC 24.
The review of interlock safety system design concludes that the safety portions of the interlock systems comply with the requirements of ANSI/IEEE Std 279. Therefore, the review finds that the interlock safety systems satisfy the requirements of 10CFR50.55a(h).

The review of the interlock safety systems examined the dependence of this system on the availability of essential auxiliary systems. Based on this review and the assumed satisfactory results of essential auxiliary supporting (EAS) systems review in other Chapters of the PSAR, the review concludes that the design of the interlock systems is compatible with the functional requirements of EAS systems.

The review of software development plans and the inspections of the computer development process and design outputs concludes that the computer systems meet the guidance of Reg. Guide 1.152. Therefore, the special characteristics of computer systems have been adequately addressed, and the review finds that the interlock systems satisfy the requirements of 10CFR50.55a(a)(1), GDC 1, 13, and 19.

7.7 Control Systems (PSAR Section 7.7)

7.7.1 Introduction

The control systems were reviewed to ensure that they conform to the acceptance criteria and guidelines, that the controlled variables can be maintained within prescribed operating ranges, and that effects of operation or failure of these systems are bounded by the accident analyses in Chapter 15 of the SAR.

The review encompassed those control systems used for normal operation that are not relied upon to perform safety functions following anticipated operational occurrences or accidents, but those that control plant processes having a significant impact on plant safety. These control systems include the systems that can, through normal operation, system failure or inadvertent operation, affect the performance of critical safety functions.

The acceptance criteria and guidelines applicable to control systems identified in SRP Section 7.7 were used for the review.

7.7.2 Control Systems Description

PSAR Section 7.7 describes instrumentation and controls of systems which are not essential for the safety of the plant except for the safety-related portions of the Neutron Monitoring System (NMS), and permits an understanding of the way the reactor and important subsystems are controlled, and why failure of these systems does not impair safety functions. The systems include the following:

1) Main Steam System (MS) - Reactor Vessel Instrumentation

   The MS sensors used for safety-related systems, engineered safeguards, and control systems are addressed in other portions of this document. The purpose of the non-safety-related MS instrumentation is to monitor and provide control input for operation variables during plant operation. The non-safety-related

instruments and systems are used to provide the operator with information during normal plant operation, or provide control for non-safety-related functions.

The MS instruments are designed to operate under normal and peak operating conditions of system pressures and ambient pressures and temperatures and are classified as non-safety-related. The non-safety-related MS instruments are powered from the non-Class 1E instrument buses.

2) Rod Control and Information System (RCIS)

The main objective of the Rod Control and Information System (RCIS) is to provide the capability to control the fine motion control rod drive (FMCRD) motors of the Control Rod Drive System (CRD) to permit changes in core reactivity so that reactor power level and power distribution can be controlled.

RCIS utilizes a dual-redundant architecture of two independent channels for normal monitoring of control rod positions and executing normal control rod movement commands. Under normal conditions, each channel receives separate input signals and both channels perform the same functions. The outputs of the two channels are continuously compared. For normal functions of enforcing and monitoring control rod positions and emergency rod insertion, the outputs of the two channels must be in agreement. Any sustained disagreement between the two channels would result in a rod block. However, when the conditions for generating a rod block signal in a single channel are satisfied, that channel alone (and independent of the other) can issue a rod block signal. For the FMCRD emergency insertion functions (Scram-Follow, ARI, SCRRI), 3-out-of-3 logic is used in the inverter control logic with the additional input signal coming from the associated emergency rod insertion panels. An automatic single channel bypass feature (only activated when an emergency insertion function is activated) is also provided to assure high availability for the emergency insertion functions when a single channel failure condition exists.

Failure or malfunction of RCIS has no impact on the hydraulic scram function of CRD. The circuitry for normal insertion and withdrawal of control rods in RCIS is completely independent of the RPS circuitry controlling the scram valves. This separation of the RPS scram and RCIS normal rod control functions prevents failure in the RCIS circuitry from affecting the scram circuitry.

3) Recirculation Flow Control System (RFC)

The objective of the Recirculation Flow Control System (RFC) is to control reactor power level, over a limited range, by controlling the flowrate of the reactor core recirculation.

The RFC consists of fault-tolerant digital controller (FTDC), adjustable speed drives (ASDs), power transformers, motor-generator (M-G) sets, switches, sensors, and alarm devices provided for operational manipulation of the ten reactor internal pumps (RIPs) and the surveillance of associated equipment. Recirculation flow control is achieved either by manual operation or by automatic operation. The reactor internal pumps can be driven to operate anywhere between approximately 30% to 100% of rated speed with the variable voltage, variable frequency power source supplied by the ASDs. When a RIP is started, the speed of the pump is slowly raised (i.e. within about 10 seconds) from a low initial starting speed to the normal minimum operating speed of approximately 30% of rated.

This system is a power generation system and is classified as not required for safety.

4) Feedwater Control System (FWC)

The Feedwater Control System (FWC) controls the flow of feedwater into the reactor pressure vessel to maintain the water level in the vessel within predetermined limits during all normal plant operating modes, including startup. The range of water level is based on steam separators performance and the need to limit carryover and carry under. Moisture carryover in the main steam impacts turbine performance while steam carry under into the downcomer region impacts Reactor Internal Pump (RIP) operation.

The FWC may operate in either single or three-element control mode. At feedwater and steam flow rates below 25% of rated (when steam flow is either negligible or else measurement is below scale), the FWC utilizes only water level measurement in the single-element control mode. When steam flow is negligible, the Reactor Water Cleanup System (RWCU) dump valve flow can be controlled by the FWC in single-element mode in order to counter the effects of density changes during heatup and purge flows into the reactor. At higher flow rates, the FWC in three-element control mode uses water level, main steamline flow, and main feedwater line flow measurements for water level control.

The FWC is a power generation control system with operation range between high water level (L8) and low water level (L2) trip setpoints. It is classified as non-safety-related. The triply redundant FWC digital controllers and process measurement equipment are powered by non-Class 1E redundant uninterrubtible power supplies (UPS). No single power failure will result in the loss of any FWC function.

5) Process Computer System (PCS)

The Process Computer System (PCS) is also referred to as the Plant Computer System in design documents as it will consist of a distributed network of computers. PCS includes three major subsystems, the Performance Monitoring and Control Subsystem (PCMS), the Power Generation Control Subsystem (PGCS) and the Plant Configuration Database Subsystem (PCDS). The PCS perform the process monitoring and control and the calculations that are necessary for the effective evaluation of normal and emergency power plant operation. The PCS is designed for high reliability utilizing redundant, network combined processing equipment which is capable of processing data, servicing subsystems, providing supervisory control over digital control systems and presenting data to the user. The PCS is non-safety-related.

6) Neutron Monitoring System (NMS)

The Neutron Monitoring System (NMS) consists of various safety-related subsystems: Startup Range Neutron Monitor (SRNM), Local Power Range Monitor (LPRM), Average Power Range Monitor (APRM), and Oscillation Power Range Monitor (OPRM) subsystems. The LPRM, OPRM, and the APRM, together, are also called the Power Range Neutron Monitor (PRNM).

The purpose of the NMS is to monitor power generation and, for the safety function part of the NMS, to provide trip signals to the Reactor Protection System (RPS) to initiate reactor scram under excessive neutron flux (and power) increase condition (high level) or neutron flux fast rising (short period) condition. The NMS also provides power information for operation and control of the reactor to the Plant Computer System (PCS), Rod Control and Information System (RCIS), Recirculation Flow Control System (RFC), and Automatic Power Regulator System (APR). The SRNM and PRNM (includes LPRM, OPRM, and APRM) Subsystems provide a safety function, and have been designed to meet the applicable design criteria.
The safety design basis for the Neutron Monitoring System is described below. All other control systems have no safety design bases. However, they are designed so that the functional capabilities of safety-related systems are not precluded.

Design features of the SRNM include:

a. The Startup Range Neutron Monitoring (SRNM) subsystem is designed to generate a high neutron flux trip signal or a short period trip signal that can be used to initiate scram in time to prevent fuel damage resulting from anticipated or abnormal operational transients.
b. The SRNM Subsystem and its preamplifier will be qualified to operate under accident and abnormal environmental conditions.
c. The independence and redundancy incorporated in the SRNM functional design will be consistent with the safety design basis of the RPS.

The design of the local power range monitor (LPRM) subsystem is to provide a sufficient number of LPRM signals to satisfy the average power range monitor (APRM) and oscillation power range monitor (OPRM) safety design bases.

The design of the APRM is such that, under the worst permitted input LPRM bypass conditions, the APRM subsystem will be capable of generating a trip signal in response to average neutron flux increases in time to prevent fuel damage. The APRM generated trip functions with trip inputs to the RPS also include: simulated thermal power trip, APRM inoperative trip, and core flow rapid decrease trip. The independence and redundancy incorporated into the design of the APRM subsystem will be consistent with the safety design bases of the RPS.

The flow rate unit, as part of the APRM subsystem, converts the core plate differential pressure signal from the Main Steam System (MS) into a core flow signal. This flow signal provides the control and reference signal for the APRM and multi-channel rod block monitor (MRBM) core flow-rate dependent trips.

The oscillation power range monitor (OPRM) is designed to provide a trip in order to prevent core flux oscillations from leading to a violation of core thermal limits. The OPRM does this while simultaneously discriminating against false signals from other parameter fluctuations that are not related to core instability.

The automated traversing probe (ATIP) subsystem portion of the NMS is non-safety related and not addressed herein.

The multi-channel rod block monitor (MRBM) subsystem portion of the NMS is non-safety-related and not addressed herein.

7) Automatic Power Regulator System (APR)

The primary objective of the Automatic Power Regulator (APR) System is to control reactor power during reactor startup, power generation, and reactor shutdown, by using appropriate commands to change rod positions, or to change reactor circulation flow. The secondary objective of the APR System is to control the pressure regulator setpoint (or turbine bypass valve position) during the reactor heatup and depressurization (e.g., to control the reactor cooldown rate). The APR System consists of redundant process controllers. Automatic power regulation is achieved by appropriate control algorithms for different phases of the reactor operation which include approach to criticality, heatup, reactor power increase, automatic load following, reactor power decrease, and reactor depressurization and cooldown. The APR System receives input from the PCS, NMS, RCIS, FWC, the PGCS, the Steam Bypass and Pressure Control (SBPC) System, and the operator's control console. APR sends output demand signals to

the RCIS to position the control rods, to the RFC System to change reactor coolant recirculation flow, and to the SBPC System for automatic load following operations. The PGS performs the overall balance of plant startup, power operation, and shutdown functions. The APR System performs only those functions associated with reactor control systems. The APR is classified as power generation system and is not required for safety.

8) Steam Bypass and Pressure Control System (SBPC)

The primary objective of the Steam Bypass and Pressure Control System (SBPC) is to control reactor vessel pressure during plant startup, power generation and shutdown modes of operation. This is accomplished through control of the turbine control and/or steam bypass valves, such that susceptibility to reactor trip, turbine-generator trip, main steam isolation and safety/relief valve opening is minimized.

SBPC generates command signals for the steam bypass valves and pressure regulation demand signals (used by the Turbine Electro-Hydraulic Control System (EHC) to generate command signals for the turbine control valves) using a triplicated fault-tolerant digital controller (FTDC). These SBPC generated signals are based on feedback signals from vessel pressure sensors. For normal operation, the turbine control valves regulate steam pressure. If, however, the total steam flow demand from the pressure controller exceeds the effective turbine control valve steam flow demand, the SBPC sends the excess steam flow directly to the main condenser, through the steam bypass valves.

Ability of the plant to follow grid-system load demands is enabled by adjusting reactor power level. This is accomplished by varying reactor recirculation flow (manually or automatically), or by moving control rods (manually or automatically). In response to the resulting steam production changes, the SBPC adjusts demand signals being sent to the EHC so that the EHC will adjust the turbine control valves to accept the steam output change, thereby controlling steam pressure. In addition, when the reactor is automatically following grid-system load demands, the SBPC permits an immediate steam flow response to fast changes in load demand. The SBPC is a power generation system and is non-safety related.

9) Fuel Pool Cooling and Cleanup System (FPCU)

The FPCU is non-safety-related. Instrumentation and control is supplied to monitor and control the fuel pool temperature. The instrumentation is for plant equipment protection.

The FPCU operates continuously during all plant modes. Evaporative and leakage losses in the system water inventory are replaced by water from the

Condensate Storage and Transfer (CSTF) System. Suppression Pool Cleanup (SPC) and Residual Heat Removal (RHR) systems also provide backup sources of make-up water to the fuel pool. If the heat load should become excessive, the RHR is operated in parallel with FPCU to remove the excess heat load when the reactor is in shutdown condition.

10) Auxiliary Fuel Pool Cooling and Cleanup System (AFPC)

The AFPC System is non-safety-related. Instrumentation and control is supplied to monitor and control the auxiliary fuel pool temperature. The instrumentation is for plant equipment protection.

The AFPC operates continuously in all plant modes. Evaporative and leakage losses in the system water inventory are replaced by water from the Condensate Storage and Transfer (CSTF) System.

Table 7.1-1 (Table 7.1-2 of the PSAR) identifies the non-safety-related control systems and the associated codes and standards applied in accordance with Section 7.7 of the Standard Review Plan.

7.7.3 Evaluation Findings

The review concludes that the design of the control systems is acceptable and meets the relevant requirements of General Design Criteria (GDC) 1, 13, 19, and 24, and 10CFR50.34(f), 10CFR50.55a(a)(1), and 10CFR50.55a(h).

A review of these systems was made for conformance to the guidelines in the regulatory guides and industry codes and standards applicable to these systems. The review concluded that the PSAR adequately classified and identified the guidelines applicable to these systems. The review finds that the control systems are appropriately designed and are of sufficient quality to minimize the potential for challenges to safety systems. The review of the system design for conformance to the guidelines finds that there is reasonable assurance that the systems fully conform to the guidelines applicable to these systems. Therefore the review finds that the requirements of GDC 1 and 10CFR50.55a(a)(1) have been met.

The review concludes that the control systems are capable of maintaining system variables within prescribed operating limits. Therefore, the review finds that the control systems satisfy this aspect of the requirements of GDC 13.

The review of control systems considered the features of these systems for both manual and automatic control of the process systems. The review finds that the features for manual and automatic control facilitate the capability to maintain plant variables within prescribed operating limits. The review finds that the control systems permit actions to be taken to operate the plant safely during normal operation, including anticipated

operational occurrences, and therefore the control systems satisfy the requirements of GDC 19 with regard to normal plant operations.

The control systems are appropriately isolated from safety systems. Therefore, the review concludes that the isolation of these systems from safety systems satisfies the requirements of 10CFR50.55a(h) and the requirements of GDC 24.

Based on the review of the defense-in-depth and diversity analysis and the quality of control system functions credited in this analysis, the review concludes that the control system complies with the criteria for defense against common-mode failure in digital instrumentation and control systems. Therefore, the review finds that the control system functions credited as diverse means for performing safety functions satisfy the requirements of Item II.Q of the USNRC Requirements Memorandum on SECY-93-087.

7.8 Diverse Instrumentation and Control Systems (PSAR Section 7.8)

7.8.1 Introduction

The objectives of this review were to ensure that the anticipated transient without scram (ATWS) mitigation systems and equipment are designed in accordance with the requirements of 10CRF50.62, and that other diverse I&C systems comply with the guidance of the NRC position on defense-in-depth and diversity (D-in-D&D) defined in SECY-93-087.

The review encompassed the diverse I&C systems and equipment provided for the express purpose of protecting against potential common-mode failures of protection systems. The following systems were covered:

- ATWS mitigation systems required for compliance with 10CFR50.62
- Diverse manual control and displays provided to comply with the NRC position on D-in-D&D as described in the Staff Requirements Memorandum (SRM) regarding SEC-93-087
- Diverse actuation systems (DAS) provided solely for the purpose of meeting the NRC position on D-in-D&D

Acceptance criteria and guidelines applicable to diverse I&C systems identified in SRP Section 7.8 were used for the review.

7.8.2 Diverse Instrumentation and Control Systems Description

The diverse I&C features of Lungmen NPS are provided for protection against common mode failures of the protection systems. These features mitigate anticipated transient without scram (ATWS) events and ensure compliance with defense-in-depth requirements. Mitigation of common mode failures is provided by the following diverse features:

1) Manual scram and MSIV isolation by the operator in the MCR in response to diverse parameter indications.
2) Core makeup water capability from the diverse feedwater, CRD, and condensate systems.
3) Availability of manual high pressure injection capability.
4) Long term shutdown capability provided in a conventionally hardwired, 2-division, analog remote shutdown system. Local displays of process variables in RSD are continuously powered and so are available for monitoring at any time.

Random failures are mitigated by the divisional sensor channel and output trip channel bypass capability of SSLC. A bypass places the remaining divisions in a 2-out-of-3 coincident logic condition such that another failure in a remaining division will not disable system operation.

7.8.2.1 ATWS Mitigation

The Anticipated Transient Without Scram (ATWS) mitigation functions use the following diverse control logics from the primary protection system but are not necessarily hardwired:

1) Alternate Rod Insertion (ARI) in association with the Rod Control and Information System (RCIS).
2) Fine Motion Control Rod Drive (FMCRD) run-in (i.e., FMCRD Emergency Insertion).
3) Automatic Depressurization System (ADS) inhibit.
4) Automatic Standby Liquid Control System (SLCS) initiation.
5) Feedwater Control System (FWC) runback.

The safety-related portions of ATWS mitigation logic, which provide an alternate means of safe shutdown, are contained within the four divisions of SSLC, but as part of circuitry that is separate and diverse from the software-based RPS logic. Analog trip modules (ATMs), instead of Digital Trip Modules (DTMs), perform setpoint comparisons for the automatic trip parameters in each division. Hardware-based discrete digital logic substitutes for software-based trip logic to perform 2-out-of-4 voting logic. SSLC evaluates manual control switch inputs for the safety systems and generates the control signal for manual backup initiation of the SSLC ATWS mitigation features.

There is a discrete-logic (non-microprocessor based) ATWS logic processor in each of the four SSLC cabinets. The ATWS logic processors are separate and diverse from RPS and other SSLC circuitry. These devices provide voting logic, control logic, and time

delays for evaluating the plant conditions for automatic initiation of SLCS boron injection and FWC runback as part of the ATWS mitigation requirement.

For ATWS mitigation, the ADS has an automatic and manual inhibit of the automatic ADS initiation. Automatic initiation of ADS is inhibited unless there is a coincident low reactor water level signal and an average power range monitors (APRMs) ATWS permissive signal (i.e., APRM signal above a specified setpoint) from the Neutron Monitoring System. There are main control room switches for the manual inhibit of automatic initiation of ADS.

The RFC System also has the logic to generate signals to mitigate an ATWS event. The alternate rod insertion function (ARI) function is non-safety-related.

7.8.2.2 Diverse Manual Controls and Displays

Diverse equipment can take the form of digital devices, digital software-based devices, or non-digital devices that are not subject to the same common mode failure as the primary protection system components. Many functions are hardwired (not multiplexed) from sensor or control switch to actuator, with diverse equipment and diverse control logic (if needed) from that of the primary protection system.

7.8.2.3 Evaluation Findings

A review of diverse I&C systems for conformance to the guidelines in the regulatory guides and industry codes and standards applicable to these systems was made. The review concluded that the guidelines applicable to these systems are adequately identified. Based upon the review of the system design for conformance to the guidelines, the review finds that there is reasonable assurance that the systems fully conform to the guidelines applicable to these systems. Therefore, the review finds that these requirements of General Design Criteria (GDC) 1 and 10CFR50.55a(a)(1) have been met.

The diverse I&C systems are appropriately isolated from safety systems. Therefore, the review concludes that the isolation of these systems from safety systems satisfies the requirements of 10CFR50.55a(h) and the requirements of GDC 24.

Based on the review of diverse I&C system status information, manual initiation capabilities, and provisions to support safe shutdown, the review concludes that information is provided to monitor the system over the anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety. Appropriate controls are provided for manual initiation of diverse I&C functions. The diverse I&C systems are designed to appropriately support actions to operate the nuclear power unit

safety under normal conditions and to maintain it in a safe condition under accident conditions. Therefore, the review finds that the design of the diverse I&C systems satisfies the requirements of GDC 13 and 19.

Based on the review of software development plans, the review concludes that the computer systems are well designed to meet the guidance of Reg. Guide 1.152. Therefore, the special characteristics of computer systems have been adequately addressed, and the review finds that the diverse I&C systems satisfy these requirements of GDC 1.

Based on the review of these functions and the design bases, the review concludes that the ATWS mitigation design includes an appropriate set of functions. Review of the interfaces of the ATWS mitigation system and equipment with the RTS concludes that the separation and independence of the RTS is not compromised by the ATWS mitigation system design. Based upon these items, the review concludes that the design of the ATWS mitigation system is acceptable and satisfies the specific design requirements identified in 10CFR50.62 for a BWR.

Review of the design bases concludes that the manual controls and displays are acceptable, independent and diverse from the safety computer system, and sufficient for manual, system-level actuation of critical safety functions and monitoring of parameters that support the safety functions. Therefore, the review concludes that the manual controls and displays fulfill the requirements of the USNRC Staff Requirements Memorandum (SRM) on SECY 93-087 item II.Q.

Review of the design bases concludes that the Diverse Manual Controls and Displays are acceptable. The functional requirements, independence requirements, and diversity requirements for these fulfill the applicable requirements of the SRM on SECY-93-087, item II.Q.

7.9 Data Communication Systems (PSAR Section 7.9)

7.9.1 Introduction

The objectives of the review of the data communications systems (DCSs) were to confirm that DCSs (1) conformed to applicable acceptance criteria and guidelines, (2) are designed to perform the safety functions assigned to them, (3) are designed to meet the reliability and availability goals assumed for the system, and (4) are designed to tolerate the effects of random transmission failures. The concern with these systems that the transmission of multiple signals over a single path may constitute a single point of failure was addressed. Lungmen DCSs covered by the review included multiplexers and general communication systems.

Acceptance criteria for the DCS were based on the acceptance criteria for the system(s) supported by the DCS. The acceptance criteria for a specific DCS include those criteria applicable to the systems supported by that DCS. These criteria are provided in SRP Section 7.9.

7.9.2 Data Communication Systems Description

The data communication network transmits signals between systems and between components of systems. Both safety-related and non-safety-related functions of the Multiplexing System are provided.

The Multiplexing System (MUX) includes both Essential and Non-Essential Multiplexing System functions. The MUX provides redundant and distributed control and instrumentation data communication networks that support monitoring and control of interfacing reactor, turbine and balance of plant systems.

The Non-Essential Multiplexing System (NEMS) is separate and distinct from the Essential Multiplexing System (EMS), though both are similar in function except for system interfaces and quality assurance requirements unique to Class 1E systems. The NEMS is diverse in both hardware and software from the EMS.

The NEMS provides distributed control and instrumentation data communication networks to support the monitoring and control of interfacing plant power generation (non-safety-related) systems. The NEMS provides all the electrical devices and circuitry, such as multiplexing units, data transmission lines and transmission controllers, to establish communications between sensors, display devices, controllers and actuators, as defined by non-safety plant system designers. The NEMS also includes the associated data acquisition and communication software required to support its function of transmitting plant-wide data for distributed control and monitoring.

The NEMS acquires both analog and digital signals from remote process sensors and discrete monitors located within a plant and multiplexes the signals to non-safety-related controllers in a central control area for control function processing. Equipment status data is transmitted to operator control panels for monitoring alarm annunciation and to the plant computer systems for data recording and display. Output control signals are multiplexed to actuators, valves, motor drives and other control equipment in the plant associated with non-safety-related systems.

Consistent with fault-tolerant digital control systems utilized in feedwater control, reactor recirculation flow control and steam bypass and pressure regulation, the NEMS is duplicated for these system interfaces, dual or triply redundant as appropriate, each with its own independent control.

The Non-Essential Multiplexing System (NEMS) portion of MUX has no safety related functions.

The EMS is a data collection and data distribution system that provides four independent distributed data acquisition and control networks to serve the four divisions of plant protection and safety systems and safety-related display systems. There are no interconnections among the four multiplexed divisions of EMS, thereby preserving divisional independence. EMS uses a deterministic communications protocol, i.e., sensor signals and control data are guaranteed network access on an equal basis without interference from other signals or network traffic. The EMS provides all of the electrical devices and circuitry, such as multiplexing units, data transmission line and transmission controllers, to establish communications between sensors, display devices, controllers and actuators, as defined by plant safety system designers. The EMS also includes the associated data acquisition and communication software required to support its function of transmitting plant-wide data for distributed control and monitoring.

The EMS acquires both analog and digital signals from remote process sensors and discrete monitors located within the plant and multiplexes the signals by redundant fiber optic lines to the MCR for transmission to the Reactor Protection System (RPS), Safety System Logic & Control (SSLC), and other divisional controllers.

The EMS receives command and control signals from SSLC and other divisional controllers in the MCR and transmits the signals by redundant fiber optic lines to safety-related RMUs in remote locations. The remote RMUs distribute the signals to the final actuating devices of the interfacing systems.

The EMS portion of the multiplexer system is designed to acquire process measurements and equipment status data, condition and format the data and then transmit the data to the MCR area. The data processed for each plant safety system in the Safety System Logic and Control system (SSLC) and other Class 1E controllers are then transmitted, via EMS, to distribution points where the signals are hardwired for use by device drivers (e.g., motor control centers (MCCs), switchgear, etc.). The EMS portion of the Multiplexer System is classified as Class 1E safety-related.

7.9.2.3 Evaluation Findings

A review of the DCS systems was made for conformance to the guidelines in the regulatory guides and industry codes and standards applicable to these systems. The review concluded that the PSAR adequately identified the guidelines applicable to these systems. Based upon the review of the system design for conformance to the guidelines, the review finds that there is reasonable assurance that the systems fully

conform to the guidelines applicable to these systems. Therefore, the review finds that the requirements of GDC 1 and 10CFR50.55a(a)(1) have been met.

The review included the identification of those systems and components for the DCS which are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. The review concludes that the PSAR has identified those systems and components consistent with the design bases for those systems. Qualification programs covered in other parts of the PSAR are assumed to adequately demonstrate the capability of these systems and components to survive these events. Therefore, the review finds that the identification of these systems and components satisfies the requirements of GDC 2 and 4.

The review of system functions concludes that the DCS conforms to the design basis requirements of IEEE Std 279 and 10CFR50.34(f). Setpoint analyses account for measurement inaccuracies attributable to the DCS in accordance with the guidance of Draft Reg. Guide 1.105. The review concludes that the DCS adequately supports RTS and ESFAS functions as necessary to sense accident conditions and anticipated operational occurrences in order to initiate protective actions consistent with the accident analysis presented in Chapter 15 of the SAR and evaluated in the SER. Therefore, the review finds that the DCS appropriately supports RTS and ESFAS compliance with the requirements of GDC 20.

The DCS conforms to the guidelines for periodic testing in Reg. Guide 1.22 and Reg. Guide 1.118. The bypassed and inoperable status indication conforms to the guidelines of Reg. Guide 1.47. The DCS conforms to the guidelines on the application for the single-failure criterion in IEEE Std 379 as supplemented by Reg. Guide 1.53. The review concludes that the DCS satisfies the requirement of IEEE Std 279 with regard to the system reliability and testability. Therefore, the review finds that the DCS satisfies these requirements of GDC 21.

The review of software development plans concludes that the computer systems conform to the guidance of Reg. Guide 1.152. Therefore, the special characteristics of computer systems have been adequately addressed and the review finds that the DCS satisfies these requirements of GDC 21.

DCS functions were included in the review of defense-in-depth and diversity analysis for RTS and ESFAS. This review concludes that the protection systems, including the DCS functions, comply with the criteria for defense against common-mode failure in digital I&C systems. Therefore, the review finds that adequate diversity and defense against common-mode failure has been provided to satisfy the diversity requirements of GDC 22 and the USNRC Requirements Memorandum on SECY-93-087, item II.Q.

The review concludes that the DCS satisfies the requirement for independence from the effects of natural phenomena. The DCS conforms to the guidelines in Reg. Guide 1.75 for protection system independence. The review of system independence and separation concludes that the DCS satisfies the requirement of IEEE Std 279 with regard to systems independence. Therefore, the review finds that the DCS satisfies the requirements of GDC 22.

The review concludes that the protection systems, including the DCS, are designed to fail into a safe mode if a condition such as disconnection of the system, loss of energy, or postulated adverse environment is experienced. Therefore, the review finds that the DCS satisfies the requirements of GDC 23.

The review of the interfaces between the DCS and plant operating control systems concluded that the system satisfies the requirements of IEEE Std 279 with regard to control and protection system interactions. Therefore, the review finds that the DCS satisfies the requirements of GDC 24.

The review of all the above concludes that the DCS satisfies the requirements of GDC 29.

The review's conclusions noted above are based upon the requirements of IEEE Std 279 with respect to the design of the DCS. Therefore, the review finds that the DCS satisfies the requirement of 10CFR50.55a(h) with regard to IEEE Std 279.

The review concludes that DCSs used in the safe shutdown systems, information systems important to safety, and interlock systems important to safety, taken in context with other provisions of the design, transmit the variables and commands necessary to maintain the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems within prescribed operating ranges during plant shutdown. Therefore, the review finds that the DCS employed by the safe shutdown systems, information systems important to safety, or interlock systems important to safety satisfy the requirements of GDC 13 and the USNRC Staff Requirements Memorandum on SECY-93-087.

DCSs have been provided to support instruments and controls within the control room to allow actions to be taken to maintain the nuclear power unit in a safe condition during shutdown, including shutdown following an accident. Equipment at appropriate locations outside the control room is also supported by the DCS design to achieve (1) prompt, hot shutdown of the reactor, and (2) subsequent cold shutdown of the reactor. Therefore, the review concludes that the DCSs employed by the safe shutdown systems, information systems important to safety, and interlock systems important to safety satisfy the requirements of GDC 19.

The review of DCS performance and diversity between the DCSs that support ATWS mitigation functions and the DCSs that support RTS functions finds that the DCS meets the requirements of 10CFR50.62.

The review concludes that DCSs used in the diverse I&C system, taken in context with other provisions of the design, transmit the variables and commands necessary to maintain the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems within prescribed operating ranges during plant shutdown. Therefore, the review finds that the DCSs employed by the diverse I&C system satisfy the requirements of GDC 13.

DCSs have been provided to support instruments and controls within the control room to allow actions to be taken to maintain the nuclear power unit in a safe condition during shutdown, including shutdown following an accident. Equipment at appropriate locations outside the control room is also supported by the DCS design to achieve (1) prompt, hot shutdown of the reactor, and (2) subsequent cold shutdown of the reactor. Therefore, the review concludes that the DCSs employed by the diverse I&C system satisfy the requirements of GDC 19.

7.10 References

1) Lungmen Units 1 & 2 Preliminary Safety Analysis Report, Taiwan Power Company.

2) NUREG 0800, "Standard Review Plan," Chapter 7 - 1997, USNRC.

3) NUREG 1503, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," May 1997, USNRC.

4) Simon, B. and Wu, Y. "Application of ANSI/IEEE Std 7.4.3.2-1993 to the Digital Protection System of the Lungmen Power Plant."

5) Facsimile Transmission from R. T. Sun (ROC-AEC) to D. A. Freed (MPR) dated April 17, 1998

5

SER on Lungmen Appendix 9A, "Fire Hazards Analysis"

9A FIRE HAZARDS ANALYSIS

Fire protection requirements for nuclear power plants are established in General Design Criteria 3 and 5 in Appendix A to 10CFR Part 50, 10CFR50.48, and Appendix R to 10CFR Part 50. Branch Technical Position BTP CMEB 9.5-1 (NUREG 0800 Section 9.5.1) incorporates the key provisions of Appendix R, Sections III.G, III.J, and III.O. The U. S. Nuclear Regulatory Commission (USNRC) issued supplemental guidance on fire protection in Generic Letters 81-12 and 86-10. In addition, the USNRC staff concluded that additional fire protection criteria were needed for the ABWR (the Lungmen design). As described in USNRC SECY 90-016, "Evolutionary Light Water Reactor Certification Issues and their Relationship to Current Regulatory Requirements" dated January 12, 1990, these additional criteria include:

> The design "must comply with 10CFR50 Appendix R, Section III.G.1.a and ensure that safe shutdown can be achieved assuming that all equipment in any one fire area will be rendered inoperable by fire and that reentry into the fire area for repairs or operator actions is not possible."
>
> The design "must ensure that smoke, hot gases, or fire suppressant will not migrate into other fire areas to the extent that could adversely affect safe-shutdown capabilities including operator actions. The control room is excluded because an alternative shutdown capability is provided which is physically and electrically independent of the control room."
>
> The design "must provide fire protection for redundant shutdown systems in the reactor containment building that will ensure, to the extent practical, that one shutdown division will be free of fire damage. Because of unique design layout, other areas may be accepted on an individual basis."

The ROC-AEC staff evaluated the Fire Hazards Analysis (FHA) for Lungmen to the same criteria used by the USNRC staff for the ABWR. Namely, the staff used the criteria of Standard Review Plan Section 9.5.1 (BTP CMEB 9.5-1, Rev. 2) and the above proposed additional criteria. Specifically, the ROC-AEC staff evaluated whether the criteria and approach for the Lungmen fire hazards analysis and preliminary design documented in the PSAR:

1. Considered potential in situ and transient fire hazards,

2. Determined the consequences of fire in any location in the plant on the ability to safely shutdown the reactor or on the ability to minimize and control the release of radioactivity to the environment,

3. Specified measures for fire prevention, fire detection, fire suppression, and fire containment, and alternative shutdown capability as required for each area containing systems, structures, and components important to safety,

4. Considered fires involving facilities shared between the two units and fires due to man-made site-related events that have a reasonable probability of occurring and affecting both units.

5. Ensured that the design complies with 10CFR50 Appendix R, Section III.G.1.a and that safe shutdown can be achieved assuming that all equipment in any one fire area will be rendered inoperable by fire and that reentry into the fire area for repairs or operator actions is not possible,

6. Considered the potential for the migration of smoke, hot gases, or fire suppressant into other fire areas and the impact of this migration on safe shutdown capabilities including operator actions, and

7. Specified redundant shutdown systems in the reactor containment with fire protection to ensure, to the extent practical, that one shutdown division be free of fire damage.

9A.1 GENERAL EVALUATION OF FIRE HAZARDS ANALYSIS

A detailed FHA is not included in the PSAR Section 9A. Instead, PSAR Sections 9.5.1.4 and 9A provide a summary review of the implementation of the Fire Protection requirements and a discussion of key features, including FHA criteria. The fire protection key features and FHA criteria were reviewed to assure the ROC-AEC fire protection criteria are met. The ROC-AEC fire protection criteria are found in:

- NUREG 0800, including BTP CMEB 9.5-1.
- 10CFR50.48 and Appendix R.
- 10CFR50, Appendix A, General Design Criteria 3 and 5.
- Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release".
- USNRC SECY 90-016, "Evolutionary Light Water Reactor Certification Issues and their Relationship to Current Regulatory Requirements".

The review also verified that the Lungmen PSAR approach for the Fire Hazards Analysis followed the approach used in the GE ABWR Fire Hazards Analysis reviewed and found acceptable by the USNRC in NUREG 1503, "Final Safety Evaluation Report Related to the Certification of the ABWR Design."

Because a detailed FHA is not included in the PSAR, a room by room review of the FHA was not performed. Instead, the criteria and approach to be used for the FHA and the applicable design drawings were reviewed to confirm overall completeness and reasonableness. The special cases, where plant features are known to deviate from the regulatory requirements, and the justifications for these deviations were reviewed in detail to assure an equivalent level of protection is provided. The review of the special cases is documented in Section 9A.2.

The ROC-AEC staff requested information from TPC in Question 9A.10 (Reference 1) regarding the qualifications of the preparer of the Lungmen Fire Hazards Analysis. TPC responded that the preparers will be qualified fire protection and reactor systems engineers per BTP CMEB 9.5-1, Position C.1.b. The Fire Protection Engineer qualifications are provided in PSAR Section 9.5.1.6.1.

9A.1.1 FIRE HAZARDS

BTP CMEB 9.5-1, Position C.1.b requires the identification of hazards and appropriate protection in locations where safety-related losses can occur as a result of:

- Concentrations of combustible contents, both in situ and transient, and continuity of the combustible contents in configurations conducive to fire spread.
- Exposure fire, heat, smoke, or water exposure that may necessitate evacuation from areas that are required to be attended for safe shutdown.
- Fire in control rooms or other locations having critical safety-related functions.
- Lack of adequate access or smoke removal facilities that impede fire extinguishment in safety-related areas.
- Lack of explosion-prevention measures.
- Loss of electric power or control circuits.
- Inadvertent operation of fire suppression systems.

The combustible contents of fire areas have not been determined at this stage of the design. Criteria to minimize the combustible loads and continuity of the combustible contents are identified in PSAR Section 9A.2.4. Also, PSAR Section 9.5.1.1.4 lists maximum allowable combustible loadings for areas of the plant without an automatic fire suppression system, for electrical equipment rooms, and for transient combustibles. The staff notes that control of transient combustible loads as well as for temporary ignition sources (e.g., hot work and welding) depend on the administrative procedures in the fire protection program. These procedures will be reviewed with the final FHA to verify implementation in the analysis.

Actual fire loadings based on the types, quantities, and characteristics of all combustibles within a fire area will be evaluated in the FHA according to PSAR Section 9A.3.1.

The staff found the above criteria and approach for evaluating combustible loads an acceptable basis for this portion of the Fire Hazards Analysis.

Smoke removal is accomplished using the HVAC system in the smoke removal mode. This mode results in the fire area with the fire being at a lower pressure than bordering fire areas such that any air leakage is into the space with the fire. This prevents migration of smoke and hot gases into bordering fire areas, which may hamper firefighting activity or safe shutdown activity. According to PSAR Section 9.4.5, smoke removal from the Reactor Building secondary containment is accomplished by the ventilation system on a divisional basis and not a fire area basis. This results in the potential for smoke migration into adjacent fire areas related to the division of the fire area with the fire. The potential for this to impede fire extinguishment is not addressed in the PSAR. The ROC-AEC staff requested information from TPC in Question 9A.6

(Reference 1) regarding the approach to be used in the FHA to consider this potential for smoke migration on the potential to impede fire extinguishment.

TPC responded, "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."

The staff finds this approach in addressing smoke migration hazards in the FHA to be acceptable.

As identified above, BTP CMEB 9.5-1, Position C.1.b requires that explosion hazards to safety related equipment be identified and appropriate protection provided. The PSAR contains no criteria regarding protection against explosion hazards. The ROC-AEC staff requested information from TPC in Question 9A.8 (Reference 1) that areas in the plant that pose an explosion hazard be identified and that the criteria and approach to protect safety-related equipment from the hazard be described. TPC responded that explosion hazards and measures taken to prevent any impact on safe shutdown equipment that are identified during detailed design activities will be considered in the FHA. At this stage in the design, no explosion hazards (except due to hydrogen buildup in containment following a design bases accident) have been identified. The staff finds this approach in addressing explosion hazards in the FHA to be acceptable.

BTP CMEB 9.5-1, Position C.1.b requires hazards to safety-related equipment associated with inadvertent operation of fire suppression systems be identified and appropriate protection provided. Criteria regarding protection against inadvertent operation of fire suppression systems have also not been provided in the PSAR, except that postulated breaks in fire protection system piping will meet the criteria of 10CFR50 Appendix A, GDC 4, as described in PSAR Section 3.6, including flooding effects. However, the possible migration of fire suppressants, e.g., water, to other areas and possible damage to safety-related equipment (e.g., damage to electrical equipment) are not addressed. The Lungmen design currently has few automatic suppression systems to assess for inadvertent operation. Also, the approach for the FHA discussed in PSAR Section 9A.3.1 includes discussing "design provisions for protecting the functional capability of safety-related systems and associated cabling from the results of inadvertent operation, careless operation, or rupture of the extinguishing systems..." The formal FHA will identify areas with this hazard and the protection provided. The staff finds this approach in addressing the hazards of inadvertent operation of suppression systems to be acceptable.

9A.1.2 SAFE SHUTDOWN CAPABILITY AND CONTROL OF RADIOACTIVE RELEASES

9A.1.2.1 Safety-Related/Safe Shutdown Equipment

To review the consequences of fire on the ability to safely shut down the plant, the safety-related equipment and safe shutdown equipment need to be identified and located. Core cooling systems required when the NSSS is isolated from the main condenser during shutdown or accident conditions are identified in PSAR Section 9A.2.5. The PSAR also includes figures 9A.4-1 through 9A.4-21 which show the fire areas, room numbers, and locations of the significant safety related system components in the Reactor Building, Control Building, and Turbine Building. These drawings do not show valve locations, instrumentation, or cable routings. A full review to identify all components important for safe shutdown in a fire area could not be performed due to the limited scope of these drawings. No detailed listing of equipment or fire area inventories were provided. These will be provided in the formal FHA that will be referenced in the FSAR. A review of the FHA drawings indicates that the major components of redundant divisions appear to be separated by 3 hour fire barriers. Further, it appears that cables can be routed such that redundant divisions are separated by 3 hour fire barriers, with some exceptions discussed in Section 9A.2. The use of the multiplexing system appears to reduce the potential for spurious actuations. Remote shutdown panels are provided in the Reactor Building to provide alternate shutdown capability for a fire in the control room. PSAR Section 9.5.1.4 commits the formal FHA to demonstrate that multiple high impedance faults will not negatively impact other equipment fed from the same power source. Thus, the criteria used in the plant design and layout appear to limit fire damage to one train of safe shutdown systems. A full review of the protection of all components important for safe shutdown in the fire areas will be performed by the staff during the review of the Lungmen FSAR.

However, structures, systems, and components (SSC) important to safety in buildings other than the Reactor Building, Control Building, and Turbine Building are not addressed as required by BTP CMEB 9.5-1, Position C.1.b. For example, fire hazards related to the areas containing components (including electrical circuits) of the Reactor Building Service Water (RBSW) system that are located outside the Control Building are not addressed but need to be addressed. The RBSW is identified in PSAR Section 9A.2.5 as a core cooling system. The ROC-AEC staff requested the following information from TPC in Question 9A.1 (Reference 1):

- Identification of the areas containing RBSW components,
- Description of the criteria for protection of these components against fire,
- Description of the criteria for the consequences of a fire in areas containing RBSW components,
- Description of the criteria for the consequences of inadvertent operation or rupture of a fire suppression system in areas containing RBSW components, and
- Identification of other systems with components located outside the Reactor Building, Control Building, and Turbine Building that fall under this CMEB position.

TPC responded that "The 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. The details of the final analysis [of the RBSW pump house] will be described in the final FHA and summarized in the FSAR." Further, TPC indicated the Fire Hazards Analysis will also address the Fire pump house, Auxiliary Building, Fuel Building, Switchgear Building, Access Control Building, Switchyard, and miscellaneous yard structures. The staff finds this approach for addressing the RBSW equipment and other systems outside the Reactor Building, Control Building, Turbine Building, and Radwaste Building to be acceptable for the FHA.

The approach for performing the FHA provided in PSAR Section 9A.3.1 includes for each fire area, identification of safety-related and non-safety-related systems and associated cabling, an analysis defining the consequences of the fire in the fire area, an analysis of the fire area addressing the consequences of fire if the fire protection system functions as designed, and design provisions to protect safety-related systems from the results of inadvertent operation, careless operation, or rupture of the extinguishing system. Provided this approach is applied to areas with safety-related or safe shutdown equipment outside the Reactor Building and Control Building, the staff finds this approach for evaluating the consequences of a fire to be acceptable for the FHA.

9A.1.2.2 Radioactive Material

To review the consequences of fire on the ability to minimize and control the release of radioactivity, areas that contain radioactive material need to be identified and located. Sources of radioactive material that can be released as a result of a fire have not been identified. PSAR Section 9A.3.1 indicates that the FHA will identify fire areas containing radioactive material which could be released. The acceptance criteria listed in PSAR Section 9A.2.4 includes the provision of ventilating systems that use filtration systems to minimize the release of radioactive materials. However, the smoke removal mode for the Reactor Building Secondary Containment HVAC system bypasses the exhaust filters. In addition, the inadvertent operation of the exhaust filter bypasses may result in a release of radioactive material. The ROC-AEC staff requested information from TPC in Question 9A.5 (Reference 1) regarding the approach to be used in the FHA to evaluate the potential for and consequences of smoke-borne radioactive releases and of radioactive releases due to inadvertent actuations of the exhaust filter bypasses.

TPC responded, "The FHA will evaluate the potential for and consequences of smoke-borne radioactive releases by compiling the combustible load data for all areas of the Reactor Building and Control Building. 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."

The staff finds the approach to evaluate the consequences of a fire on the control of radioactive releases in the FHA to be acceptable.

9A.1.3 FIRE PROTECTION FEATURES

9A.1.3.1 Fire Barriers

According to the analysis approach for the formal FHA, PSAR Section 9A.3.1, redundant safety-related system divisions are separated by 3 hour fire barriers in the Reactor Building and Control Building, with the exception of primary containment, the Main Control Room, and isolated special cases. PSAR Section 9.5.1.3.11 discusses these exceptions. In primary containment, the redundant divisions will be separated by as much distance as possible and the containment will be inerted so that a fire is not possible. For the control room, a remote shutdown system is provided in another fire area as an alternate safe shutdown capability. According to PSAR Sections 9.5.1.1.3 and 9A.2.4, doors, electrical cable firestops, piping penetration firestops, and HVAC divisional wall penetration fire dampers are to have a fire rating equivalent to the barrier they penetrate, or are to be analyzed for equivalent fire rating. Further, structural steel members are fireproofed to the required fire rating by application of UL-listed or FM-approved materials or boxing designs.

PSAR Section 9A.3.4 provides justification for not installing fire dampers in the four Atmospheric Control System supply and exhaust lines for the wetwell and drywell. These lines contain containment isolation valves that are normally closed except during plant outages. If a fire occurs during a plant outage, smoke removal would be accomplished and drywell and wetwell sprays would be initiated to keep the containment at a temperature well below the threshold of damage for the duct. This exception is acceptable based on the provided justification.

PSAR Section 9A.3.5 provides descriptions of two wall designs that either are UL tested and approved as a 3-hour fire barrier or will be tested per UL. Provided these walls are designed, tested as appropriate, and installed to maintain the fire rating, these wall designs are considered acceptable.

PSAR Section 9A.3.6 indicates some fire doors may deviate from BTP CMEB 9.5-1, Position C.5.a(5). Specifically, fire doors with other functions (e.g., watertight, pressure, or missile protection) that are specified to rated and labeled fire door criteria will be identified as rated fire doors. Fire doors that require the fire rating label to be deleted per other criteria will be identified as an equivalent to the fire rating. Doors that are not constructed as a fire door will be identified by its main function. These deviations are acceptable, provided that the required fire resistive rating for the door can be demonstrated and is documented.

Based on these criteria, the staff finds this approach to fire barriers to be acceptable.

9A.1.3.2 Fire Detection System

According to PSAR Sections 9.5.1.3.7 and 9.5.1.3.9, all areas of the plant are provided with ionization product of combustion detectors or thermistor wire heat detection equipment. Detectors are designed in accordance with NFPA 72 Class A and NFPA 70 Class 1 requirements. The EDG rooms utilize thermal and infrared cross-zoned detection systems to actuate an automatic foam sprinkler extinguishing system. Two actuation signals are required to initiate the system.

According to PSAR Section 9.5.1, fire detectors are not installed inside consoles and cabinets in the Control Room Complex as recommended by BTP CMEB 9.5-1, Position C.7.b. The justification for this position is based on:

- Continuous manning of the Control Room Complex.
- Limited combustibles inside the consoles and cabinets.
- Consoles and cabinets designed to be cooled by air so that smoke will exhaust into the Control Room Complex.
- Fire in any single cabinet or console will not disable safe shutdown capability.

The staff finds this exception to be adequately justified for the FHA, provided these conditions are verified in the formal FHA.

Manual fire alarm stations are provided as required by the ROC Fire Protection Code.

The staff finds the above approach for fire detection to be acceptable, provided the FHA is specific regarding the type and coverage of fire detectors used.

9A.1.3.3 Fire Suppression Systems

According to PSAR Section 9A.2.4, criteria for the fire suppression systems evaluated in the FHA include:

- Automatic wet pipe sprinkler systems for areas in which transient fire loading will most likely occur for normal maintenance operations.
- At least two fire suppression means are available to each fire area, including:
  - -- Automatic sprinkler
  - -- Automatic foam water spray
  - -- Standpipe and hose reels
  - -- Class ABC hand extinguishers.

- Standpipe and hose reel systems will be supplied with water in the event of a safe shutdown earthquake or any single failure mode. The effects of pipe breaks in fire suppression systems meet the criteria of 10CFR50 Appendix A, GDC 4, as described in PSAR Section 3.6.

Fire suppression requirements are listed in PSAR Sections 9.5.1.1, 9.5.1.3, and 9.5.1.4 and meet the requirements from BTP CMEB 9.5-1, Position C.1.c and C.6.c. Areas other than the EDG rooms and EDG Day Tank rooms that may require automatic suppression as the primary coverage have not been identified. The FHA criteria documented in PSAR Section 9A.2 do not identify requirements for automatic suppression systems to meet BTP CMEB 9.5-1. For example, BTP Position C.5.e(2) requires automatic suppression for safety-related cable trays that may be affected by a potential exposure fire unless:

(1) the number of equivalent standard 24 inch wide cable trays in a given fire area is six or less,
(2) the cabling does not provide instrumentation, control, or power to systems required to achieve and maintain hot shutdown, AND
(3) smoke detectors are provided in the area and continuous line-type heat detectors are provided in the cable tray.

According to PSAR Section 9B.2.3.3, calculations to establish the maximum allowable combustible loadings for areas without an automatic suppression system assumed 30 inch wide cable trays. In accordance with BTP CMEB 9.5-1, Position C.5.e(2), a 30 inch wide cable tray is equivalent to two 24 inch wide cable trays when determining the need for automatic suppression for safety-related cable trays. As a result, an area with a stack of cable trays may require automatic suppression based on the BTP position but would not require automatic suppression based on the combustible loading calculations. These conditions would require deviations from the BTP.

Deviations from the BTP need to be identified in the formal Fire Hazards Analysis and adequate justification provided. The staff finds the above criteria and requirements for fire suppression to be adequate for the FHA, provided the fire suppression requirements from BTP CMEB 9.5-1 are met or the deviations identified with appropriate justification.

9A.1.3.4 Alternate Shutdown Capability

As required by BTP CMEB 9.5-1, Position C.1.d, alternative shutdown capability is provided for fires in the Control Room Complex. According to PSAR Section 9.5.1.3.11, remote shutdown panels are provided in a fire area different than for the Control Room Complex. Further, PSAR Section 9.5.1.1.2 identifies features of the remote shutdown panels. The approach for the FHA discussed in PSAR Section 9A.3.1 indicates the design criteria used in providing fire protection for the equipment within each fire area will be identified in the FHA. This includes the criteria to address the lack of divisional separation within the Control Room Complex. No other areas requiring alternate shutdown were identified for the FHA. The staff finds this approach to evaluating alternate shutdown capability to be acceptable.

9A.1.4 SHARED FACILITIES AND SITE-RELATED EVENTS

BTP CMEB 9.5-1, Position C.1.b requires that fires involving facilities shared between units and man-made site-related events that could affect more than one reactor unit be considered. The

FHA criteria described in PSAR Section 9A.2 and the FHA approach described in PSAR Section 9A.3 do not address this CMEB requirement. Three questions were raised:

- Impact of fire in facility containing the swing diesel generator.
- Migration of smoke from external fires into safety-related buildings.
- Impact of construction hazards at Unit 2 during Unit 1 operation.

One shared building, the Auxiliary Fuel Building, houses a swing diesel generator capable of connecting to the electrical systems of either of the two Lungmen units. Although this diesel has not been identified as safe shutdown equipment, it has circuits associated with the electrical systems of both units. The ROC-AEC staff requested information from TPC in Question 9A.2 (Reference 1) regarding the criteria and approach to be used in the FHA to evaluate the impact of fires in shared facilities on safe shutdown of both units be described, with particular emphasis on the swing diesel. The staff also requested a description of the criteria for fire protection features for these facilities. TPC responded that "The swing diesel generator electrical configuration uses IEEE 384 separation methods to preclude the possibility of a common mode failure." The staff finds this approach to address this question to be acceptable.

Exterior walls of safety-related buildings are designated 3 hour fire barriers but contain no fire dampers in HVAC system supply ducts according to PSAR Section 9.5.1.1.6. As a result, smoke from an external fire may be drawn into safety-related buildings (The Control Room Complex HVAC system is an exception since it isolates on detection of smoke in the supply duct), hampering shutdown activities and possibly activating various smoke alarms. The ROC-AEC staff requested information from TPC in Question 9A.4 (Reference 1) regarding the criteria and approach to be used in the FHA to evaluate the impact of external hazards, including smoke migration, on safe shutdown.

TPC responded that "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. 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."

TPC also stated, "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."

The staff reviewed the TPC response and finds this approach to be generally acceptable for this question. However, the staff has identified one open item regarding the Reactor Building secondary containment HVAC system. This system provides ventilation air to areas for all three safety divisions. Smoke from an external fire may potentially be distributed to safety-related

areas for all three divisions, resulting in alarms from smoke detectors in each of these areas. This could result in operator confusion. Also, to remove smoke that migrated into each divisional area, operation in the smoke removal mode may not be possible until the smoke from the external fire clears from the ventilation system inlets. The staff requested further information from TPC (Reference 2) regarding a description of design features that minimize the impact of this common mode scenario for the RB secondary containment HVAC system, including the type of damper to be used and whether smoke detectors would be included in the supply ducts. This is considered an open item.

The third question concerns the analysis of construction hazards on safe shutdown. 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. The ROC-AEC staff requested information from TPC in Question 9A.3 (Reference 1) regarding the criteria and approach to be used in the FHA to evaluate hazards for the operating unit due to construction activities at the other unit be discussed. This discussion was to include the configuration of the fire protection water supply system during the construction period. TPC responded that "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." The staff finds this approach to address this question to be acceptable.

9A.1.5 COMPLIANCE WITH 10CFR50 APPENDIX R, SECTION III.G.1.A

PSAR Section 9.5.1.4 specifies that the FHA assumes the functional loss of all equipment, including electrical cable, located within the fire area experiencing the fire. The analysis further assumes that redundant equipment out of the fire area is available and capable of performing the required safety or shutdown function. These assumptions, if verified, satisfy the first criterion from USNRC SECY 90-016.

9A.1.6 MIGRATION OF SMOKE, HOT GASES, OR FIRE SUPPRESSANT

Hazards associated with smoke and hot gas migration have been addressed in other sections of this SER (e.g., sections 9A.1.1 and 9A.1.4).

PSAR Section 9A does not address the potential for the migration of fire suppressant to other fire areas. Migration of fire suppressant to other fire areas may result in flooding of the other fire areas and potential damage to additional safety-related equipment. PSAR Section 9A.3.1 only indicates that the consequences of pipe breaks are addressed in PSAR Section 3.6. Areas without fire protection piping but may be subject to manual fire suppression would not be addressed in PSAR Section 3.6. The approach for the FHA should include an analysis addressing the potential for and consequences of migration of fire suppressant to other fire areas. This is considered an open item.

9A.1.7 REACTOR CONTAINMENT FIRE PROTECTION

The criterion from USNRC SECY 90-016 that the design must provide redundant shutdown systems in the reactor containment with fire protection to ensure one shutdown division be free of fire damage. As discussed above, it appears that the FHA will verify the plant design will meet this requirement for the secondary containment. The primary containment is considered one fire area and contains redundant safe shutdown systems within it. PSAR Section 9.5.1.1.2 provides justification for not dividing the primary containment into three fire areas. The justification is based on inerting the containment during power operation, alternative fire suppression methods, limiting active safe shutdown components inside containment to ADS/SRV valves and check valves, and maintaining maximum separation between divisional equipment. Specific cases are discussed in Section 9A.2 below. The staff finds the approach for the FHA addresses this requirement.

9A.2 SPECIAL CASES

These special cases identify known deviations from the BTP CMEB 9.5-1 requirements. They are identified in PSAR Section 9A.5. Each was reviewed in detail to verify that an equivalent level of protection is provided. This is acceptable to the staff

9A.2.1 REACTOR BUILDING PIPING PENETRATIONS

In the PSAR, TPC commits to providing Reactor Building piping penetration details that demonstrate a 3-hour fire resistance in the FSAR. The staff finds this approach to providing an equivalent level of protection acceptable.

9A.2.2 FIRE DOOR DEVIATIONS

Fire doors with other overriding design criteria (e.g., the Reactor Building airlock doors) may be of special construction. Such doors will generally have a backup fire door. All of the special construction doors with a backup fire door are not identified at this time. However, the staff finds this approach to providing an equivalent level of protection acceptable.

9A.2.3 CHARCOAL FILTERS FOR PROCESS TANKS AND DRAIN SUMPS

Several tanks and sumps may have small charcoal canister-type filters for their vents. According to PSAR Section 9A.5.3, these filters will not be provided with temperature monitoring or fixed suppression systems, but that manual suppression systems will be available at or near each filter. No justification for this exception is provided. The ROC-AEC staff requested information from TPC in Question 9A.7 (Reference 1) regarding the contribution of these filters to an exposure fire to safety-related equipment, the fire detection or fire containment systems to be used, and the justification for the stated position. TPC responded, "The requested information is part of the longer term design process. At this time, no charcoal canister-type filters ... 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." The staff finds this approach for the FHA to be acceptable.

9A.2.4 PIPE BREAK ANALYSIS

High pressure fire water systems require analysis as high energy and moderate energy lines per PSAR Section 3.6. PSAR Section 3.6 addresses the effects of high and moderate energy piping breaks and cracks inside and outside containment, including the effects of pipe whip, jet impingement, spraying, and flooding. The staff finds this approach to be acceptable.

9A.2.5 FIRE SEPARATION FOR DIVISIONAL SYSTEMS

Areas with cables from more than one division in close proximity require special justification. These areas include:

1. RPS Scram Circuits
2. Main Steamline Radiation Monitor Detectors
3. Main Steamline ADS Relief Valves
4. Main Steamline Isolation Valve Control and Limit Switch Interfaces
5. Under the Reactor Vessel
6. Local Instrumentation and Control Equipment
7. Leak Detection and Isolation System
8. Standby Liquid Control
9. Flammability Control System
10. Standby Gas Treatment System
11. Fine Motion Control Rod Drive Motors
12. Reactor Building Operating Deck Radiation Monitors
13. RHR Shutdown Cooling Outboard Isolation Valves
14. RCIC System
15. Containment Isolation Valves
16. Division 4 Sensors

Justification for the existence of more than one division of electrical cables in close proximity for these systems and components relies primarily on one or more of the following:

- Running cables in separate conduits,
- Providing fuse and other circuit protection,
- Providing maximum spatial separation possible,
- Providing barriers within enclosures,
- Relying on redundant components to achieve system function.

For the RHR shutdown cooling outboard isolation valves, the RCIC Main Steam supply outboard isolation valve, and outboard containment isolation valves in Division 2 and 3 areas, electrical circuits for these valves are associated with a safety division different than the safety division of the line in which the valve is installed. As a result, these electrical circuits are in fire areas containing safety-related/shutdown equipment of a different safety division. The justification provided in PSAR Section 9A.5.5 discusses only the capability of achieving the valve's function if damaged by fire. However, no discussion is provided regarding the protection of upstream

power supplies from hot shorts or shorts to ground, nor the potential for propagating the fire to a fire area of another division via this circuit. As a result, a fire in this fire area may prevent operation of multiple safety divisions. PSAR Section 9A.5.7 presents an analysis of typical circuits of special cases, but excludes the above special cases. The ROC-AEC staff requested information from TPC in Question 9A.9 (Reference 1) information regarding the protection of these electrical circuits. TPC responded, "Dedicated conduit raceways will be used in all cases to ensure availability of these containment isolation valves during the postulated fire scenarios." The staff finds this approach for preventing shorts as a result of exposure to fire to be acceptable for these cases.

9A.2.6 TYPICAL CIRCUITS ANALYSIS OF SPECIAL CASES

For typical cases where a device from one division is located in an area of another division, PSAR Section 9A.5.7 with Figure 9A.5-2 and Table 9A.5-2 summarizes the method of protecting the electrical circuit and justifies the acceptability of the loss of function of the device. For these circuits, Regulatory Guide 1.75 and IEEE-384 are met. The staff finds this approach to protect these circuits acceptable.

SUMMARY

The staff finds the criteria and approach used for the Lungmen Fire Hazards Analysis acceptable. The FHA criteria and approach described in PSAR Section 9A conform to the applicable sections of BTP CMEB BTP 9.5-1 and the additional criteria from USNRC SECY 90-016, except for two open items. These open items are considered minor issues that do not need to be resolved prior to issuance of a construction permit but should be resolved prior to the issuance of an operating license. The two open items are:

1. Smoke Migration into the RBSC HVAC System from External Hazards (SER Section 9A.1.4). Describe the RB secondary containment HVAC system design features that minimize the impact of smoke migration from external fires on all three safety divisions within the RB secondary containment.

2. Migration of Fire Suppressant (SER Section 9A.1.6). Provide the FHA approach to analyze the potential for and consequences of migration of fire suppressant (other than due to a pipe break) to other fire areas.

6

SER on Lungmen PSAR Chapter 19, "Severe Accident Analysis"

19. SEVERE ACCIDENT ANALYSIS

19.1 Introduction

The purpose of this section is (1) to describe the approach used in the review of the severe accident analysis presented in the PSAR prepared for the Lungmen NPS and (2) to evaluate the approach proposed and analysis performed by TPC and its contractors for resolution of severe accident issues for the Lungmen NPS.

The approach adopted in this evaluation considers the U.S. Nuclear Regulatory Commission's (USNRC's) policy statement on severe reactor accidents regarding future designs and existing plants (Reference 19-1) and its approach to resolution of severe accident issues for evolutionary light water reactors (LWRs) as specified in SECY-90-016 (Reference 19-2), SECY-91-262 (Reference 19-3), SECY-93-087 (Reference 19-4), and the corresponding staff requirements memorandums (SRMs). Additional considerations related to the unique perspective of severe accident risk at the Lungmen NPS have also been included in establishing criteria for the evaluation of severe accident analysis performed in the Lungmen PSAR Chapter 19.

Furthermore, the type of attributes and criteria specified in the draft Standard Review Plan Chapter 19 issued by the USNRC (Reference 19-5) have been used in this review for general guidance. Even though this draft document was prepared for the risk-informed decisionmaking on plant-specific changes to an operating plant's current licensing basis, the same type of considerations (e.g., defense-in-depth, safety margin, current regulation, etc.) is equally applicable to the application of construction permit or operating license for a new plant.

Severe accidents are those accidents more severe than design basis accidents (DBAs) in which substantial damage is done to the reactor core whether or not there are serious offsite consequences. To provide adequate protection of the public health and safety, current regulations in the nuclear industry require conservatism in design, construction, testing, operation, and maintenance of nuclear power plants. A defense-in-depth

approach has been mandated in order to prevent accidents from happening and, if accidents should occur, to mitigate their consequences. Siting of nuclear power plants in less populated areas is emphasized. Furthermore, emergency response capabilities that provide additional defense-in-depth protection to the surrounding population are also mandated by the pertinent government agencies.

The reactor and containment systems design provides a vital link in the defense-in-depth philosophy. Existing reactors and containments are designed to withstand a LOCA and to meet the siting criteria of 10 CFR Part 100 and General Design Criteria (GDC) of 10 CFR Part 50, Appendix A. The large-break LOCA and other accidents analyzed in accordance with the USNRC's Standard Review Plan (NUREG-0800) and documented in Chapter 15 of the Lungmen PSAR are commonly referred to as "design-basis accidents" for nuclear power plants. This high level of confidence in a defense-in-depth approach results, in part, from stringent requirements for meeting single-failure criterion, redundancy, diversity, quality assurance, and utilization of conservative models.

In the U.S., the regulatory agency also has requirements to mitigate adverse conditions associated with transients or events considered outside the design basis such as anticipated transient without scram (ATWS) (10 CFR 50.62; Reference 19-6), station blackout (SBO) (10 CFR 50.63; Reference 19-7), and combustible gas control (10 CFR 50.44; Reference 19-8); however, a definitive set of regulatory requirements for addressing specific severe accident phenomenon or issue does not exist. This is partly because the focus of the USNRC prior to the accident at Three Mile Island Unit 2 (TMI-2) was on design basis accidents. However, more recently, the USNRC has required licensees and applicants to evaluate their plants for severe accident vulnerabilities. For almost 20 years since the TMI-2 accident, no new plants have been ordered in the United States. In the absence of new construction permit applications, no clear requirements in the area of severe accident analysis, other than the completion of a probabilistic risk assessment, have been developed for the application of new plants. Current regulations that require conservative analyses and inclusion of mitigative features for design-basis events provide margin for severe accident challenges. To ensure that the Lungmen NPS design achieves a higher standard of severe accident safety performance, it was concluded that severe accidents should be considered as part of the PSAR during the design stage.

For the evaluation of the severe accident analysis submitted in the Lungmen PSAR, criteria specified for resolution of severe accident issues in SECY-90-016 and SECY-93-087 are considered as applicable to the Lungmen facility. Furthermore, additional considerations related to site specific hazards, and effects of the differences between the Lungmen NPS and the certified, Standard Advanced Boiling Water Reactor (ABWR) have also been incorporated into the determination of acceptability for severe accident concerns. The following sections describe the criteria that were used for the evaluation of severe accident issues.

The evaluation of severe accident analysis discussed in this portion of the report includes only the results of a review of PSAR Chapter 19 prepared for the Lungmen NPS. The scope of the severe accident analysis review does not include any other sections of the PSAR (e.g., Appendices to the Lungmen PSAR). Because the Lungmen NPS is very similar to the Standard ABWR reviewed by the USNRC, some of the discussions in this chapter have been taken directly from NUREG-1503 (Reference 19-9). In all such cases, however, the applicability of the NUREG-1503 descriptions and conclusions to the Lungmen NPS has been verified.

The objectives of the review of severe accident analysis are to:

- Evaluate whether the analysis performed in support of the design and construction of the Lungmen plant is adequate and reasonable in terms of the scope, completeness, methods, assumptions, data, etc.,
- Determine if the Lungmen design satisfies the applicable industry and regulatory requirements in terms of severe accident prevention and mitigation.
- Assess whether the overall results are reasonable and consistent with previous experience,
- Examine the balance and adequacy of plant design in preventing and mitigating severe accidents,
- Measure against the various safety-related goals established by industry and in the ABWR Standard Plant licensing review bases,

- Identify strengths and weaknesses of the Lungmen plant design features

The major aspects of this review of the severe accident analysis includes an assessment of the plant capability for severe accident prevention and mitigation, an evaluation of the containment performance analysis, an examination of the severe accident management plan, and an investigation of the containment pressure capacity evaluation included in the Lungmen PSAR.

19.2 Assessment of Severe Accident Prevention

Accidents can be initiated from internal and external hazards. The initiation of accidents from the internal hazards can be represented by a set of accident initiators. Generally, the external hazards can also lead to the same initiators with some additional impact on plant equipment. Accident initiators can be separated into two general groups: transients and LOCAs. Transients include planned reactor shutdowns and transients that result in reactor scrams. Examples of transients include manual shutdown, main steam isolation valve closure, loss of condenser vacuum, loss of feedwater, nonisolation event (trip with bypass), inadvertent open relief valve, and loss of offsite power. LOCAs generally fall within three categories, small, medium, and large, based on the size of the line break.

Following the accident initiator, plant systems respond to control reactivity, reactor pressure, reactor water level, and containment parameters within the design basis spectrum. Ensuring sufficient heat removal from the core to prevent overheating and subsequent fuel damage is of paramount importance. Failure to provide this heat removal can result in fuel overheating and the potential for oxidation and melting of the reactor core.

Transient-induced accidents are usually accompanied by actuation of the safety or relief valves that transfer steam from the reactor directly to the suppression pool, thus limiting an increase in containment pressure. LOCAs blow down into the drywell atmosphere until the suppression pool horizontal vents are uncovered, resulting in containment pressurization lower than the containment design pressure. In the event that the steam from the reactor bypasses the suppression pool and travels directly to the wetwell atmosphere, significant pressurization of the containment may result. The issue of

suppression pool bypass is addressed in Sections 19.3.3.5 of this report for severe accident bypass.

Four technical issues applicable to the severe accident prevention of the ABWR design were identified by the USNRC as significant to reactor safety. These issues include anticipated transients without scram, station blackout, fire protection, and intersystem LOCA (ISLOCA). They were developed after evaluation of the current operating reactor designs and experience, evolutionary and advanced light-water reactor (ALWR) designs, insights from a number of Probabilistic Risk Assessments (PRAs) of LWRs and ALWRs, etc. To resolve these technical issues, the USNRC developed criteria for evolutionary LWRs to prevent the occurrence of accident initiators from leading to a severe accident. These criteria were specified in SECY-90-16 and SECY-93-087 and include design provisions for ATWS, SBO, fires, and intersystem LOCAs.

The above technical issues and their criteria for resolution were developed for the certification of new, generic reactor designs. For application of a construction permit or an operating license at a specific site, it is necessary to evaluate the plant features that are intended for the response to and/or mitigation of both internal and external hazards. For each safety functions, conformance to the defense-in-depth philosophy as well as redundancy, independence, and diversity requirements should be addressed. Probabilistic measures of the plant's ability to prevent severe accidents (e.g., failure likelihood of safety functions, conditional probabilities of severe accidents resulting from various initiators) should be considered in demonstrating the adequacy of the plant design to prevent the occurrence of accident initiators and, ultimately, severe accidents.

The Lungmen NPS is a variation of the Standard ABWR design, which has already been certified by the USNRC. To evaluate the severe accident issues, it is important to consider the differences in design and operations between the Lungmen NPS and the Standard ABWR that are relevant to the severe accident prevention. The effects of these differences on Lungmen's ability to prevent severe accidents should be addressed.

19.2.1 General Review Approach

In the area of assessment of the capability for severe accident prevention, plant features designed to prevent the occurrence of severe accidents have been evaluated to examine the margin in plant capability both in redundancy/diversity and in performance. This includes the prevention of the occurrence of accident initiators (e.g., LOCAs and transients) and plant system response to accident initiators in maintaining safety functions (i.e., control of reactivity, reactor pressure, vessel water level, heat removal, and containment parameters).

In this review, the following principles were be examined:

- The plant design meets the current industry and regulatory requirements.
- Defense-in-depth concept is used.
- Sufficient safety margins exist.
- The conditional and unconditional risks (core damage and large early release) are acceptable and within the bounds of applicable, industry safety goals.

The appropriate industry design and regulatory criteria (e.g., redundancy, independence, and diversity) were adopted in this portion of the review as part of the evaluation criteria. These criteria were applied in the considerations of the safety functions required for the response to the various accident initiators. The applicable regulations and the associated criteria developed for the evolutionary LWR certification issues were also used in the relevant sections of this evaluation.

In the review of defense-in-depth concept, the following attributes were considered, where applicable:

- Challenges to the integrity of the barriers.
- Likelihood of failure of each barrier.
- Failure dependencies among barriers.
- Overall redundancy and diversity.

In light of the preceding, this evaluation has also given special attention to the following aspects:

- No over-reliance on programmatic activities or operator actions to compensate for weaknesses in plant design.
- Requirements for system redundancy, independence, and diversity are met, consistent with the frequency and consequences of challenges to the systems. Proper balance in these requirements should be maintained among different safety functions.
- Defenses against common cause failures and human errors are considered.

With respect to the evaluation of safety margin, where applicable, and information is available, it was based on engineering (or other pertinent) analysis and data that identify the safety margin, or plant activities conducted to preserve those margins.

Prevention of severe accidents was therefore evaluated using both deterministic and probabilistic criteria. Deterministic criteria include such considerations as regulation requirements, redundancy, independence, diversity, etc. Probabilistic criteria consider the frequency of accident initiation, conditional likelihood of core damage given accident initiators, safety function failure probabilities, etc.

Accident initiators identified in historical BWR operating experience and past PRA analyses that are applicable to the Lungmen NPS were used as a framework against which this assessment was performed. This framework was supplemented by the specific findings obtained in the PRA completed for the Lungmen NPS.

Evaluation of plant capability to prevent severe accidents given occurrence of accident initiators was based on the plant systems that are designed to maintain the safety functions in response to accident initiators. An evaluation of adequate and sufficient plant features for severe accident prevention was performed. Engineering analyses of plant response to various accident initiators was examined, where applicable and information available. These analyses include those presented in the Lungmen PSAR (including thermal-hydraulic analyses given in the accident analyses and severe accident analysis chapters) and those performed for similar plants in past studies.

Analyses performed for plants other than Lungmen were used with considerations of differences between Lungmen and these other plants.

For the probabilistic portion of the review, the frequencies of accident initiators were compared with the range of values calculated in previous, similar studies to determine if adequate design is provided to ensure that the frequencies of accident initiation are within acceptable bounds. Available, conditional frequencies of core damage for accident initiators were used to measure the effectiveness of plant features for the prevention of severe accidents given the occurrence of accident initiators. Unconditional frequencies of core damage were used in this review as an indication of the combined effects of the plant features for the prevention of accident initiator occurrence and for the prevention of severe accidents given occurrence of accident initiators.

Furthermore, available failure probabilities of systems/safety functions were used to measure the reliability performance of plant features for severe accident prevention. They were also be used to evaluate the balance of design in diversity and redundancy for various safety functions.

In addition to the accident initiators discussed in the preceding, special attention was given to the following areas due to their importance for the BWR plants:

- Anticipated Transient Without Scram
- Station Blackout
- Fire Protection
- Intersystem Loss-of-Coolant Accident

The criteria established by the USNRC for resolution of these evolutionary LWR certification issues were used in determining the acceptability of the Lungmen NPS design with respect to these severe accident concerns.

19.2.2 Severe Accident Prevention Capability

The assessment of severe accident prevention includes the following areas:

- General features for severe accident prevention

- Evolutionary LWR technical issues related to severe accident prevention
- Site specific external hazards: typhoons and earthquakes
- Differences between the Lungmen NPS and the Standard ABWR

19.2.2.1 General Features for Severe Accident Prevention

The following discusses the general features included in the Lungmen NPS for severe accident prevention.

19.2.2.1.1 Preventive and/or Mitigative Features

There are three independent safety-related, Engineered Safety Feature (ESF) divisions, any of which can safely shut down the reactor. In addition, there are non-safety-related systems (e.g., condensate and feedwater systems) that can be used to achieve safe shutdown.

The following discusses the general features for prevention of accident initiators and safety functions required for preventing the occurrence of severe accidents:

Prevention of Accident Initiators

Reduction in LOCA Frequency

The LOCA frequency is reduced as explained below:

- Due to the use of reactor internal pumps (RIPs) and elimination of the recirculation system external piping, the LOCA frequency is reduced.
- The frequency of ISLOCA is reduced because of the higher design pressure for low pressure systems that may interface with the reactor coolant pressure boundary.

Reduction in Transient-induced Scrams

Scram avoidance is achieved by the following design features:

- Fast transfer to the station main generator in the event of loss of offsite power.
- The 110% steam bypass following such event as load rejection.
- Reduction in reactor power by selected control rod run-in and trip/runback of RIPs following loss of feedwater heating, load rejection, turbine trip, and fast load winddown.

Automatic Reactor Shutdown

The redundant and diverse scram mechanisms include:

- Hydraulic rod insertion initiated by 2/4 Reactor Protection System (RPS) automatic and manual scram signals.
- Hydraulic rod insertion initiated by Alternate Rod Insertion (ARI) from 2/3 Recirculation Flow Control System (RFC) signals.
- Electromechanical drive-in of control rods initiated by 2/3 RFC signals.
- Injection of boron solution by Standby Liquid Control System (SLCS) from Safety System Logic and Control (SSLC) signals.

Vessel Pressure Control

Reactor vessel pressure control is accomplished through:

- 10 turbine bypass valves with a total capacity of 110% steam relief.
- 18 Safety/Relief Valves (SRVs) including 8 Automatic Depressurization System (ADS) valves. SRVs provide three pressure control modes; i.e., overpressure relief operation, overpressure safety operation, and ADS.
- RCIC turbine (partial capacity during the initial response following a scram).

Vessel Water Level Control

High Pressure Makeup

This function is achieved by:

- Two redundant and independent divisions of High Pressure Core Flooder (HPCF) initiated by SSLC.
- One diverse HPCF division (i.e., RCIC for core cooling) with diverse power source and hard-wired initiation circuit.
- Non-safety-related feedwater system in conjunction with condensate system.

Low Pressure Makeup

This function consists of:

- Three redundant and independent ESF divisions, each containing a low pressure core injection and a decay heat removal subsystems.
- Diverse and independent water addition for core and containment cooling by ACIWA using an independent water supply, emergency diesel generator (EDG), and control.
- Non-safety-related condensate system.

Decay Heat Removal

The decay heat removal function can be accomplished through:

- Three redundant and independent ESF divisions, each containing a decay heat removal and a low pressure core injection subsystems.
- Heat removal by non-safety-related main condenser.

- Steam relief through containment overpressure protection system (COPS) rupture disk.

Support Functions

- Three independent and separated safety-related, electrical divisions, one for each of the ESF divisions.
- Three independent and separated EDGs, one for each of the safety-related electrical divisions.
- A completely diverse Swing Diesel Generator (SDG) electric power source.
- Three redundant, independent, and 100% reactor building cooling water (RBCW)/reactor building service water (RBSW) for the removal of heat loads associated with ECCS pump operation and suppression pool heat.

19.2.2.1.2 Basis for Acceptability

For the severe accident prevention capability, the most important improvements in the Lungmen NPS design over the existing BWRs are the reduction in scram and LOCA frequencies and the use of three independent safety divisions. For most accident initiators, the high pressure core cooling capability is enhanced by the design of three HPCF divisions. Similarly, the three divisions of decay heat removal system is also a significant improvement. The elimination of the core spray system is compensated by the design of three divisions of low pressure core injection and the addition of ACIWA system. To some extent, the COPS can serve a similar containment heat removal function performed by the hardened vent at some of the existing BWRs.

Based on the discussions presented in the preceding section for the various safety functions necessary for severe accident prevention, the Lungmen NPS satisfies the defense-in-depth concept and complies with all of the regulatory criteria for redundancy, independence, and diversity. In addition, based on the generally lower frequencies of internal accident initiating events and the relatively low core damage frequency, the applicable performance requirements for the prevention of accident initiation and the

occurrence of severe accidents are met. However, to complete the evaluation of safety margin as specified in Reference 19-5, additional probabilistic measures are needed. These probabilistic measures can also be used to evaluate the balance in the plant design among various safety functions. The probabilistic measure of safety functions reliability was not addressed in Chapter 19 of the PSAR for the Lungmen NPS. In addition, Appendix A to the Lungmen PSAR provides insufficient intermediate and final results (only a limited number of top contributors to core damage are presented) to permit the use of probabilistic parameters for this purpose. No commitment has been made by TPC to provide this information in Chapter 19 or Appendix A of the revised PSAR. This remains an open item.

19.2.2.2 Evolutionary LWR Technical Issues

The technical issues identified for the certification of evolutionary LWRs that are relevant to the severe accident prevention includes ATWS, SBO, fires, and intersystem LOCAs.

19.2.2.2.1 Anticipated Transients Without Scram (ATWS)

An ATWS is an anticipated operational occurrence followed by the failure of the trip portion of the RPS. Anticipated operational occurrences (transients) are those conditions of normal operation that are expected to occur one or more times during the life of the nuclear power plant and include, but are not limited to, loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power. Dependent upon the transient and its severity, the plant may recover and continue normal operation or the plant may automatically shut down (scram) via the RPS. The RPS is designed to safely shutdown the reactor to prevent core damage.

These transients when coupled with a failure of the RPS may lead to conditions beyond the design basis of the plant. In these cases, the reactor must be manually scrammed in order to avoid reactor fuel damage or coolant system damage. Subsequent failure of the manual scram system and inadequate core cooling may lead to core damage.

Transients with the greatest potential for significant damage to the reactor core and containment are those leading to an increase in reactor pressure and temperature, a

loss of heat sink, or a failure of the RPS to scram the reactor. During an ATWS event, reactor power, pressure, and temperature must be controlled or the potential exists for a severe accident.

The ATWS rule (10 CFR 50.62) was promulgated to reduce the likelihood of an ATWS event and to enhance mitigation capability if such an event occurred. For future BWRs, the ATWS rule specifies inclusion of the following design features:

- An ARI system that is independent and diverse from the RPS.
- Automatic initiation of the SLCS with 325.5 L/min of 13 wt% sodium pentaborate solution.
- Automatic recirculation pump trip (RPT) under conditions indicative of an ATWS.

In SECY-90-016, it is recommended that diverse scram systems should be provided for evolutionary LWRs. However, the June 26, 1990 SRM indicated that, if it can be demonstrated that the consequences of an ATWS are acceptable, the demonstration can be accepted as an alternative to the diverse scram system.

19.2.2.2.1.1 Preventive and/or Mitigative Features

The Lungmen NPS has included design features to reduce the frequency of reactor scram, to enhance the capability for automatic reactor shutdown, and to mitigate the consequence of an ATWS event.

Scram Avoidance

Regarding the reduction of scram frequency (i.e., scram avoidance), improvements over the existing BWRs have been made to eliminate some of the mechanisms and causes for transients and thus reactor scrams. This is evidenced by the generally reduced frequency estimates for anticipated transients. The ability to fast transfer to the station main generator in the event of a loss of offsite power and the 110% steam bypass capacity provided by the Steam Bypass and Pressure Control System (SBPC), which serves to reduce the reactor vessel pressure increase following such events as load

rejection, will also decrease the scram frequency, since they remove some of the immediate effects of the transient events.

Scram avoidance is further accomplished by quickly and sufficiently reducing the reactor power (i.e., eliminating the need to scram) following such events as loss of feedwater heating, load rejection, turbine trip, fast load winddown. The Rod Control and Instrumentation system (RCIS) actuated by the Recirculation Flow Control System (RFC) is designed to initiate the selected control rod run-in (SCRRI) to reduce power. In addition, the RFC simultaneously reduces the recirculation flow by tripping a portion of the operating RIPs and/or by running back the RIPs to their minimum speed to ensure sufficient power reduction.

Automatic Reactor Shutdown Capability

The Lungmen design for reactor shutdown includes hydraulic control rod insertion by the Control Rod Drive System (CRD) in response to automatic and manual scram signals from the RPS, CRD hydraulic control rod insertion initiated by ARI in response to signals from RFC/RCIS, CRD electromechanical drive-in of the control rods in response to signals from RFC/RCIS, and injection of the aqueous boron solution by SLCS in response to ATWS signals from the SSLC.

Reactor Protection System

The RPS is the primary scram system for protection against anticipated transients. It consists of four divisions each with its independent sensors, multiplexing channel, digital microprocessor-based trip module (DTM), and dual scram load drivers. When 2 or more of the 4 DTMs determine a trip condition, the load drivers disrupt the electric power to the individual scram pilot valves that cause the hydraulic insertion of control rods by opening the hydraulic control unit (HCU) scram valves.

Alternate Rod Insertion

A primary backup to the RPS is provided by ARI. Control logic for manual and automatic initiation of ARI is provided by RFC. RFC is a microprocessor-based, three-division digital controller system. The ATWS actuation logic provided by the RFC is 2 out of 3.

In the event that RPS fails to trip the reactor, RFC initiates scram signals to the ARI. These signals are high reactor dome pressure, low reactor water level 2, or manual operator action. Upon receipt of any of these signals, ARI exhaust valves on the scram air header would open to depressurize the scram air header, causing the hydraulic insertion of all of the control rods in the same manner as the RPS initiated scram. The ARI function is a delayed hydraulic scram.

Fine Motion Control Rod Drive

The same RFC signals that actuate ARI simultaneously initiate motor run-in of all operable Fine Motion Control Rod Drive (FMCRD) by the RCIS. The FMCRD insertion is a slower, electromechanical control rod insertion function.

Standby Liquid Control System

A second backup reactivity control system is provided by the SLCS. It injects boron solution into the reactor pressure vessel (RPV) through the HPCF loop B discharge piping. The SLCS is initiated by the SSLC ATWS logic circuitry. The SLCS initiation signals include:

- High vessel pressure and Startup Range Neutron Monitor (SRNM) ATWS permissive for 3 minutes,
- Low vessel water level 2 and SRNM ATWS permissive for 3 minutes, or
- Manual ARI/FMCRD run-in signals and SRNM ATWS permissive for 3 minutes.

The SSLC ATWS logic circuits are separate and diverse from RPS and other SSLC circuitry. Each of the four SSLC divisions comprises process sensors, analog trip module (ATM), and a non-microprocessor logic processor. The ATM output is hardwired to the trip logic processor in each SSLC division. The actuating signals for SLCS are hardwired to the respective system controllers.

Mitigation of ATWS Consequence

In addition to providing scram signals to ARI, the RFC system also initiates recirculation pump trip (RPT) on indication of an ATWS. The actuation signal for the RPT is high vessel pressure when the RIPs are not connected to the M/G set, and low vessel water level 2 when the RIPs are connected to the M/G set. The recirculation runback will be initiated on any scram signals, or any ARI/FMCRD run-in signals. By lowering the recirculation flow, the reactor power level can be reduced. As part of the ATWS mitigation, the SSLC ATWS logic circuitry also initiates automatic runback of the Feedwater Control System (FWC) on high vessel pressure and SRNM ATWS permissive for 2 minutes. The feedwater runback and lowering of the vessel water level are designed to reduce thermal-hydraulic instability and power oscillation.

The automatic initiation of ADS is designed to be inhibited when there is a coincident low reactor water level signal (level 1.5) and an Average Power Range Monitor (APRM) ATWS permissive signal.

ATWS Rule Compliance

The ARI system and the electrical insertion of FMCRDs use sensors and logic which are diverse and independent of the RPS. Processing of the sensor output signals in the RFC is completely separate from that in the RPS. The RPS opens the HCU scram valves by actuating the scram pilot valves and dumping the air pressure to the individual HCU scram valves. The ARI, on the other hand, opens the HCU scram valves by actuating the scram air header valves and dumping the scram air header pressure. Therefore, by the use of RFC and redundant ARI scram air header exhaust valves, the ARI system provides an alternate means for actuating hydraulic rod insertion that is independent and diverse from the RPS from the sensor output to the final actuation device. The reliability of the ARI scram function is enhanced due to the diversity design and the use of 2 of 3 RFC initiation logic.

By the injection of an aqueous solution of sodium pentaborate, the SLCS initiated by the SSLC ATWS logic circuitry provides a diverse means to shut down the reactor. In addition, with the use of ATMs in the SSLC ATWS mitigation channels, which is completely diverse from the microprocessor-based RPS channels using DTMs, the SSLC ATWS logic circuits are separate and diverse from RPS circuitry. The single

failure criterion is met and the system reliability is assured by the inclusion of redundant trains. Due to the design of automatic initiation by the SSLC ATWS logic circuitry, the SLCS reliability is significantly improved over the conventional SLCS design used in the currently operating BWRs. The 325.5 L/min of sodium pentaborate solution specified in the ATWS rule for 637.5-cm RPV is satisfied by the 379 L/min provided for the 706-cm Lungmen vessel.

By eliminating the scram discharge volume and the use of electric motor-driven run-in diverse from the hydraulic insertion feature, the Lungmen RPS design has eliminated the potential for mechanical common mode failure of the existing locking-piston control rod drive.

As such, the Lungmen NPS has included all of the features required by the ATWS rule; i.e., a reliable RPS, an independent, diverse, and reliable ARI system, an automatic SLCS, and automatic RPT equipment. To further enhance the accident prevention capability, two additional features have also been included in the Lungmen design; i.e., scram avoidance and diverse electromechanical mechanism for rod insertion.

19.2.2.2.1.2 Basis for Acceptability

The Lungmen NPS has a number of design features that reduce the risk from an ATWS event including a diverse scram system with both hydraulic and electric run-in capabilities for the fine motion control rod drives (FMCRD), an automatic SLCS, and a RIP trip capability. In addition, the scram discharge volume has been removed from the Lungmen NPS design, eliminating some of the potential problems that could affect the scram function associated with older BWR designs.

The Lungmen NPS has an alternate rod insertion system that is diverse, when compared with that of the RPS, from sensor output to the final actuation device. The ARI system has redundant scram air header exhaust valves and is designed to perform its function in a reliable manner.

The core design for Lungmen Units 1 & 2 will be based on GE12 fuel design, which is different from the fuel type used in the ABWR Standard Safety Analysis Report (SSAR) evaluations. The main turbine bypass valves for Lungmen are 110% capacity as compared to 35% capacity for the Standard ABWR. The GE12 fuel design data

necessary for ATWS analyses have not been finalized at the time the PSAR was issued. As such, the transient analysis results based on the 33% bypass valve capacity and on previous GE6/7 (P8x8R) fuel data was used as a surrogate. Compared to the performance requirements for fuel integrity, containment integrity, suppression pool temperature, primary system pressure, and long term shutdown cooling, the results are found to be acceptable.

The total ATWS frequency and ATWS-induced core damage frequency are within acceptable range. The ATWS core damage frequency is dominated by the seismic events, which contribute to more than 13% of the total core damage frequency.

In SECY-90-16, the USNRC concluded that evolutionary LWR designs should provide diverse methods of inserting control rods to mitigate a potential ATWS and to ensure a safe reactor shutdown. The Lungmen NPS incorporates a diverse method for inserting control rods. The Lungmen NPS design is capable of satisfactorily mitigating the effects of an ATWS and preventing an ATWS event from evolving into a severe accident with core damage. It is concluded that the Lungmen NPS meets the criteria specified in SECY-90-016 and complies with the ATWS rule through incorporation of the features discussed above. To demonstrate that there are no vulnerabilities in the Lungmen NPS design to prevent and mitigate the ATWS event associated with external events, TPC has committed to include the ATWS frequencies and ATWS core damage frequencies in the revised Lungmen PSAR Chapter 19 for the risk important external events, in addition to internal events.

19.2.2.2.2 Station Blackout (SBO)

An SBO involves the complete loss of ac electrical power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the on-site emergency ac power system). SBO does not include the loss of available ac power to buses fed by station batteries through inverters or by alternate ac sources, nor does it assume a concurrent single failure or DBA.

During normal plant operation, power is supplied to the Class IE distribution system from the main generator or offsite grid at certain plants. Following plant shutdown, the

preferred power source is the offsite grid, which provides a continuous source of ac electric power to equipment required to maintain core coolability. If the power from the offsite grid is not available, the on-site distribution system will sense an undervoltage condition and initiate a transfer to the EDGs for continued power. In the event of the loss of both the offsite grid and EDGs, an SBO has occurred. As most DHR and containment heat removal systems are dependent upon ac power for operation, failure to provide core cooling during an SBO will likely result in core temperature and pressure increases and may lead to a severe accident.

The SBO rule (10 CFR 50.63) allows several design alternatives to ensure that a plant is able to withstand an SBO for a specified duration and recover. In SECY-90-016, it is stated that the preferred method of demonstrating compliance with 10 CFR 50.63 is through the installation of a spare (full-capacity) alternate ac power source of diverse design that is consistent with the guidance in Regulatory Guide (RG) 1.155 and is capable of powering at least one complete set of normal shutdown loads. The applicable regulation for an alternate ac source as approved by the USNRC is as follows: The standard design must provide an alternate ac power source for the purposes of dealing with station blackout.

A complete evaluation of the Lungmen NPS relative to the SBO rule is provided in following sections.

19.2.2.2.2.1 Preventive and/or Mitigative Features

Loss of Offsite Power and SBO

Due to the high likelihood of earthquakes and typhoons in the Taiwan area, the frequency of loss of offsite power may be higher than most of the nuclear plants in the U.S. The loss of normal AC power will lead to a fast transfer to the station main generator to supply house electric loads. Should the fast transfer fail, the turbine will trip and reactor will scram due to high condenser pressure on loss of circulating water. Due to this capability of fast transfer to the station main generator, the frequency of loss of offsite power initiating event should be significantly lower than the frequency of loss of offsite power.

The Lungmen NPS is designed with three independent EDGs, one for each of the three ESF divisions. Each ESF division supported by a full-capacity EDG is equipped with a complete set of safe shutdown capability. In addition to the EDGs, a completely diverse Swing Diesel Generator is also provided. It is connectable to any of the three safety divisions of either unit and is capable of powering one complete set of normal safe shutdown loads. No plant AC support systems are needed to operate the SDG. Thus, the frequency of SBO events at the Lungmen NPS is expected to be lower than plants in the U.S since the probability of losing all three EDGs plus the SDG is very small. Given a station blackout event, however, the time required to restore offsite power is expected to be longer compared to the average U.S. plants because the fraction of loss of offsite power events induced by earthquakes and typhoons is likely to be higher. This is due to the potentially more severe damage to the offsite power transmission system caused by these external events.

SBO Response - High Pressure Core Cooling Operation

Following a reactor scram caused by a loss of offsite power initiating event, the MSIVs would close, RIPs would coast down, and feedwater pumps would also coast down. The loss of feedwater will cause the RPV to isolate on low water level. Failure of the EDGs to initiate and failure of the SDG will leave the RCIC as the only source of high pressure water for core cooling.

In response to an SBO event, RCIC system will be actuated when the water level reaches Level 2 to provide the high pressure core injection function. Operators will control the RCIC to maintain the reactor vessel water level between Level 3 and Level 8. To ensure stable and continued operation of the RCIC turbine, adequate steam pressure must be available. To avoid SRV cycling, operators must control the reactor vessel pressure by manually opening and closing the SRV(s). If the operator fails to manually control pressure, the SRVs will cycle on the spring setpoint. Battery power and nitrogen supply are required for the manual operation of SRVs. Sufficient battery power from all three divisions is designed to support this operation. Nitrogen supply from the SRV accumulators and backup N_2 bottles are available as the normal supply from the atmospheric control system becomes unavailable due to loss of AC power. Use of the backup nitrogen bottles requires operator action to manually open a closed supply valve at the valve station.

Control of the SRVs to prevent increase in the vessel pressure is only necessary for the initial period of operation with the RCIC system. After this initial period, the steam drawn by the RCIC turbine is more than sufficient to prevent vessel pressure increase. The DC battery power needed for the RCIC operation is supplied from the division I battery, which is designed to support RCIC for 8 hours.

The water source for RCIC injection is provided by CST and suppression pool. RCIC is aligned to initially take suction from the CST. Its suction source will automatically switch to the suppression pool on high suppression pool level or low CST level. Since the RCIC turbine oil (cools the turbine components) is cooled by the water flowing through the RCIC pump, the suction source must be manually switched back by the operators to the CST when the suppression pool heats up to the high suppression pool alarm setpoint. This will prevent failure of the RCIC turbine as a result of excessive oil temperatures.

SBO Response – Low Pressure Core Cooling Operation

If the RCIC fails to operate during the first 8 hours in an SBO event or if the AC power is not restored within 8 hours, the operators must depressurize the reactor vessel using SRVs and initiate low-pressure injection by the ACIWA system. This system can provide low-pressure injection indefinitely as long as the SRVs are held open and reactor vessel remains depressurized.

Decay Heat

Since no AC power is available in an SBO event, capability for containment heat removal is lost. During the period the RCIC is providing the core cooling function, decay heat is dumped into the suppression pool via the SRVs and/or RCIC turbine exhaust steam. After RCIC becomes unavailable, core heat is dumped into the suppression pool through the SRVs. Without containment cooling, temperature and pressure inside the primary containment will continue to increase until the Containment Overpressure Protection System (COPS) actuates (i.e., rupture disk opens). If the rupture disks fail to open, the containment backpressure will cause the SRVs to close, the reactor vessel will repressurize, and the ACIWA injection will be lost.

19.2.2.2.2.2 Basis for Acceptability

The Lungmen NPS design includes three independent electrical divisions, each capable of providing power to a high-pressure and low-pressure water injection division, each powered by a full-capacity EDG, and each division capable of independently shutting down the reactor. In addition, the Lungmen NPS design includes an alternate source of ac power (i.e., swing DG) to back up the EDGs. The RCIC system (with its supporting systems) is designed to perform its function without ac power for 8 hours. Extended blackout capabilities are also provided by the ACIWA system. This system allows for makeup to the reactor vessel following depressurization or to the containment sprays from a direct-drive diesel fire pump or by connecting an external pumping source, such as a fire truck, to a yard standpipe into the RHR system. Sufficient nitrogen capacity is provided at the Lungmen NPS for the operation of SRVs to control reactor vessel pressure.

Due to the design of fast transfer to the station main generator on loss of normal power, the frequency of loss of offsite power initiating event is reduced. With three independent EDGs and the alternate AC source from SDG, the likelihood of station blackout is significantly decreased. The capability to mitigate the consequence of an SBO event has also been improved for the Lungmen NPS. First, the ability of the DC-controlled, steam-driven high pressure water injection source to provide core cooling has been extended to 8 hours. Second, capability to continue to supply core cooling after loss of the steam-driven high pressure injection system has been augmented due to the addition of ACIWA and COPS. The Lungmen NPS is capable of maintaining the core cooling function indefinitely following an SBO event. This is a substantial improvement over the currently operating BWRs which can only provide core cooling for as long as the battery can last (about 2 to 6 hours). For the SBO response, the only area that the currently operating BWRs has better capability than the Lungmen NPS is that there are two steam-driven high pressure systems in some existing plants for core cooling during the initial period following an SBO event.

In SECY-90-016, the USNRC concluded that designers should meet the SBO rule by including an alternate ac power source of diverse design capable of powering at least one complete set of normal shutdown loads. To cope with SBOs, the Lungmen NPS has included an SDG, a diverse source of AC power. Based on the preventive and

mitigative features described above, it is concluded that the Lungmen NPS has met the criteria of SECY-90-016 and the applicable regulation for station blackout. To demonstrate that there are no vulnerabilities in the Lungmen NPS design to prevent and mitigate the SBO event, TPC has committed to include the SBO frequencies and SBO core damage frequencies in the revised Lungmen PSAR Chapter 19 for typhoons and earthquakes, in addition to internal events.

19.2.2.2.3 Fire Protection

The USNRC concluded that fire protection issues that have been raised through operating experience and the external events program must be resolved for future, evolutionary light water reactors. In SECY-90-016, it was recommended that current NRC guidance to resolve fire protection issues be enhanced to minimize fire as a significant contributor to the likelihood of severe accidents for advanced reactors. As indicated in SECY-90-16, the advanced reactor design must ensure that safe shutdown can be achieved, assuming that all equipment in any area will be rendered inoperable by fire and that reentry into the fire area for repairs and operator actions will be impossible. Because of its physical configuration, the main control room is excluded from this approach, provided an independent alternative shutdown capability that is physically and electrically independent of the control room is included in the design. The advanced reactor design must also provide fire protection for redundant shutdown systems in the primary containment that will ensure, to the extent practicable, that one shutdown division will be free of fire damage. Additionally, the advanced reactor design must ensure that smoke, hot gases, or fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe-shutdown capabilities, including operator actions. The safety-grade designation for the fire protection systems must be such that the remaining or redundant shutdown systems are protected. These fire protection measures are for both DBAs and severe accidents.

19.2.2.2.3.1 Preventive and/or Mitigative Features

The following elements are selected features included in the Lungmen NPS design and fire protection program to prevent and mitigate fires.

Prevention of Fire Initiation

Allowable combustible loadings for both fixed and transient combustibles will be established for the Lungmen NPS to limit the amount of combustibles contained in a fire area. Combustible materials and ignition sources are to be kept to the absolute minimum. It must be ensured that their use is unavoidable. Administrative controls will also be provided to control transient combustibles, ignition source permits, etc.

The use of plastic materials, including electrical cable insulation, has been minimized in the Lungmen NPS design. Combustible loading due to cable insulation is minimized by locating the power sources adjacent to the loads served and multiplexing the control signals to and from the control room. As a result, most of the cables to and from the control room and the cable spreading rooms have been eliminated. Multiplexing is also used within the control room so that the cables between panels have been reduced to mostly power cables. The use of fiber optic cables in the multiplexing system thus reduces both ignition sources and combustibles.

Except for lubricating oil in the EDG rooms and the CRD pump room, there are no storage areas in the Reactor or Control Building for flammable liquids, oxidizing agents, flammable compressed gases, or explosive or highly flammable materials. Transformers located within fire areas containing safety-related equipment will be of the dry-type only. Small quantities of chemicals may be stored in listed or approved cabinets and containers for immediate use. Combustible liquids are required to be kept in approved containers to reduce the likelihood of spill.

Paper in Control Room Complex is required to be stored in approved containers except when in use. Only limited combustibles are contained in the control room cabinets and consoles. Above the suspended ceiling in the Control Room Complex, only cables associated with lighting and the fire alarm system are routed. Cables in this area are enclosed in conduit. The raised floor area of the Control Room Complex is used for routing of cables from the control room cabinets, panels, and the divisional electrical rooms. The amount of cabling in this area is substantially reduced over previous designs. All low voltage power cables will be contained in flexible or rigid conduit in this area. No transient combustibles will be stored in this area during normal activities. The likelihood of fire ignition in this area is very low.

The materials used inside the containment are nonflammable or flame resistant wherever practical. During normal plant operation, fires in the primary containment are prevented by nitrogen inerting.

Restricting Fire Spread and Propagation

Fire Confinement by Separation

The Lungmen NPS will be designed with three independent and physically separated safety-related divisions. Each division can safely shut down the plant. If one division becomes disabled due to a fire, there are still two independent, redundant divisions available to provide safe shutdown functions.

With the exception of the primary containment, control room complex, and remote shutdown room (when operating from the remote shutdown panels), etc., each division will be separated from its two redundant ones by 3-hour fire barriers. Control, power or instrument cables of redundant systems that are used for safe shutdown, are also separated by 3-hour fire barriers. In addition to separation of equipment/systems of different safety divisions, the walls, floors, and ceilings of the Nuclear Island Buildings have 3-hour fire resistive rating where required by a high combustible loading in the room. The fire barrier system will be designed to confine smoke, hot gases, fire suppressant, etc. to the division of fire origination. Penetration assemblies will be designed to meet the fire-resistive ratings of the barriers they penetrate. Support systems such as HVAC and RBCW are designated as safety-related if they support safety-related systems. These support systems are given divisional assignments and are separated by fire barriers.

The Nuclear Island (NI) buildings are laid out internally so that fire areas of like divisions are grouped together to the extent possible. In addition, the grouping of fire areas of like divisions is coordinated from building to building so that divisional fire areas line up adjacent of each other at the interface between the Reactor and Control Buildings.

As such, piping, HVAC ducts, and cable trays are grouped together in divisional arrangements and do not require routing of services of one division across areas of redundant divisions. The plant layout and design is such that the redundant portions of

the safety-related systems are located in different fire areas. Each fire area will only contain safe shutdown equipment of one division. Fire barriers can inhibit fire progression and contain a fire in its originating fire area. Therefore, the most serious consequence of a fire is that it may incapacitate one safety or safe shutdown division.

When cable trays are arranged in stacked arrays in the NI buildings, power cable is routed in the uppermost tray to the greatest extent possible to provide better separation from other trays in the stack.

Main Control Room (MCR)

In MCR, all four safety divisions of instrumentation and control are present. The divisional panels in the Control Room are physically separated to the extent practicable. The cabling from the divisional electrical equipment rooms will be routed to the Safety System Logic Control cabinets with divisions I and III on one side of the operator area and divisions II and IV on the opposite side. A fire in any single cabinet or console will not disable the capability to safely shut down the plant.

Cables in the area above the suspended ceiling are enclosed in conduit. Divisional cable trays in the subfloor area will be enclosed and separated horizontally by a minimum of 0.91 m.

Fire Prevention/Mitigation Inside Containment

In the primary containment, all safety divisions are present. The primary containment is inerted and protected from fire during normal operation. It is, however, de-inerted during a plant outage. The design features and administrative control programs described in the following limit the fire risk during shutdown and ensure that a fire in the containment during shutdown will not result in a loss of the ability to keep the core cool.

The materials used inside the containment are nonflammable or flame resistant wherever practical resulting in low combustible loading. Fire protection program limits transient combustibles and ignition source permits. The large volume of the containment allows for spatial separation of the three redundant, divisional components inside the primary containment. In addition, the primary containment does not contain any equipment other than check valves, piping, and ADS/SRVs that are required to operate

for safe shutdown. Reactor core cooling and decay heat removal can still be achieved even if a fire disables all of the active components inside the containment. Furthermore, the drywell spray system can be used as a sprinkler system either by itself or in conjunction with manual firefighting activities to suppress a fire in the containment. The containment purge fans can be operated in a smoke removal mode to prevent the spread of smoke outside the containment.

Fire Detection and Suppression

Fire Protection Systems (FPS)

Fire detection for all areas, except the EDG rooms, is provided by the ionization-type product of combustion systems or thermistor wire heat detection systems. The fire detection systems in the EDG rooms utilize cross-zoned heat detectors and infrared detectors to initiate an automatic foam sprinkler extinguishing system. Both manual and automatic alarm systems are provided in all areas of the plant. Fire detection and suppression alarm systems have 4-hour battery packs and are also supplied with power from a Non-Class 1E uninterruptible power supply.

Select alarm information is transmitted to the MCR. The fire alarm systems will be zoned by division according to the divisional assignment of the area that each zone covers. When a fire occurs, it is apparent to the operators which divisional area is experiencing the fire. If necessary, the operators can shut down the equipment in the affected division and transfer operations to one of the two unaffected divisions. All power supplies in the affected division can be de-energized. Unless necessary, the ventilation fans within the divisional fire area experiencing the fire would not be shut down so that they can provide smoke removal function.

In general, fixed automatic suppression will be provided in areas where allowable limits of combustible loading are exceeded. Automatic fire suppression systems are initiated on a divisional basis so that there are no interactions between divisions. Each division of automatic fire suppression system has its own dedicated detection and actuation equipment for the control of the fire sprinklers in that divisional area.

The fire suppression systems are designed such that there are at least two fire suppression systems available to each fire area. Primary suppression coverage for all buildings is provided by wet standpipes and hose reels. Secondary coverage is provided by portable ABC extinguishers. Portable fire extinguishers are provided throughout the buildings. Automatic wet pipe sprinkler systems are provided for areas in which a transient fire loading is most likely to occur as a result of combustibles introduced by normal maintenance operations. Areas covered by sprinklers or foam systems are also covered by the manual hose system. There will be two manual hose streams available (i.e., can be reached from at least two hose stations) in any area containing equipment required for safe shutdown and covered only by manual hose system. Standpipes are fed from two directions. The NI design provides the following types of suppression and utilizes them in suitable combination for the fire hazard considered:

1. Automatic sprinkler (wet pipe or preaction foam water)

2. Automatic foam water spray

3. Standpipe and hose reels

4. Class ABC hand extinguishers

The sprinkler systems and the portions of the wet standpipe system within the Control and Reactor Buildings and one train of the fire suppression water supply system will be designed to remain functional following a safe shutdown earthquake. The AC independent water addition system can be connected to the Reactor Building fire protection system header. The duration of foam discharge from the automatic foam-water extinguishing systems will be a minimum of 10 minutes. Distance from a hose reel and extinguishers is no more than 25 m from any location.

The Control Room subfloor area will be equipped with fire detection capability, but no fire suppression system. However, fire can be detected quickly and manual fire fighting can be started without delay since the MCR is continuously manned. Since fire resistant cables will be used, the amount of water needed to extinguish fire is relatively small. No fire detectors will be installed inside the cabinets and consoles in the Control Room Complex. These cabinets and consoles contain only limited combustibles and are air-

cooled. Smoke from a cabinet fire will exhaust to the Control Room Complex and be quickly detected by the Control Room operators. No automatic fire suppression systems will be installed in the office spaces in the Control Room Complex.

The Turbine Building is provided with standpipe, hose reels, and portable extinguishers throughout the building.

The main power, unit auxiliary, and reserve transformers are provided with deluge water spray suppression systems. The systems are automatically actuated by flame or temperature detectors. An oil and water collection pit is provided beneath each transformer. Drain piping will direct the oil and water from the transformer pits to oil and water separators. Shadow type fire barrier walls are provided between adjacent transformers.

Seismic Design of the FPS

The fire protection piping systems are seismically designed if their failure could affect safe shutdown systems. During normal operation, the seismically designed and nonseismically designed systems are separated by normally open motor operated valves which will isolate the seismic portion from the nonseismic portion of the fire loop such that a break in the nonseismically analyzed portion of the system cannot impair the operation of the seismically designed portion of the system. Following a safe shutdown earthquake, there will be no uncontrolled release of fire suppression water in the areas.

With the exception of the EDG rooms, the fire detection systems are not seismically qualified, as they are passive and have no control function. The EDG room fire detection system is seismically qualified to protect against inadvertent actuation.

The fire barrier system for safety-related areas is also designed to Seismic Category I.

Seismic-fire interactions are minimized with the seismic design of those FPS pipings, fire detection systems, and fire barriers that may affect the safe shutdown system.

Smoke Removal

The HVAC systems are designed for the dual purpose of HVAC and smoke control. Once a fire has been discovered, the HVAC system for the area experiencing the fire is switched to its smoke removal mode. The smoke control system maintains the areas for the two redundant divisions at a positive pressure relative to the area of division experiencing the fire. As such, migration of combustion products and thus fire propagation to areas of the redundant division will be prevented so that safe shutdown capabilities, including operator actions, would not be adversely affected.

Smoke will be removed by the normal HVAC system. In both the Reactor and Control Building, each fire area is served by its corresponding divisional HVAC system. The control room complex is served by a fourth HVAC system. In the Reactor Building, the normal supply and exhaust fans are located external to the building. A single non-safety-related HVAC System supplies normal cooling for secondary containment in the Reactor Building. Every room of the Reactor Building secondary containment receives supply air from and exhausts to the building normal HVAC system. Within the Reactor Building, the system is branched into three separate systems with smoke dampers and fire dampers for each branch. The emergency ventilation systems for the electrical equipment and diesel generator rooms provide additional smoke removal capability for those rooms.

The smoke control system will be designed so that it can

- Prevent pressure buildup due to a fire.
- Provide pressure control across the fire barriers to ensure that any leakage is into the fire area experiencing the fire
- Prevent backflow of smoke and hot gases when fire barrier doors are opened for access for manual fire suppression activities.
- Remove smoke and heat from the fire.
- Supply clean air for fire suppression personnel.

The HVAC systems in the fire areas not experiencing a fire continue their normal operation so that the pressure in the other fire areas remain at a positive value relative to the fire zone involved in the fire. Due to the relative positive pressure of the nonfire area and the operation of smoke removal mode in the fire area, combustion products are carried back into the area of the fire and out the area of the fire's exhaust duct system.

Personnel Access Routes

The personnel access routes for fire suppression activities will be compatible with the design of fire barriers, HVAC, and smoke removal. Clean, cool air will be provided for access routes to fire areas by fans outside of the fire area experiencing the fire. This and the smoke removal control allow access to the fire area for manual fire fighting.

Consequences of Fire

Safe Shutdown Equipment and Associated Cabling

Safe shutdown equipment susceptible to fire damage and its associated cabling located in a fire area can be damaged by fires in the area. The extent of damage will be limited to only the division corresponding to the fire area.

Spurious Control Actions

The use of the fiber optic cables in the multiplexing system with dual channels can essentially prevent spurious control actions that may be caused by fire-induced hot shorts. However, the Lungmen NPS still includes some conventional control and instrument cables that are susceptible to hot shorts.

With the multiplexing system, equipment will continue to operate until manually shut down in the field, if ability to control from the MCR is lost. Fire in the MCR will not prevent manual starting of the equipment at the switchgear or motor control centers.

Effects of Fire Suppressant Introduced by FPS

The Lungmen NPS is designed so that accumulation, drainage, or spray of the fire water will not affect the ability of the redundant divisions to safely shut down the plant.

Effect of Fires on Structural Steel

The structural steel members are protected from fires by application of a UL-listed or FM-approved cementicious or ablative material, or by a UL- or FM-approved boxing design.

Interactions between Buildings

No safety-related services required by the Reactor Building are provided from or through the Turbine Building. Therefore, fires in the Turbine Building will not disable the safety-related equipment in the Reactor Building. They can only affect the non-safety-related power sources supplied through the Turbine Building to the safety-related loads in the Reactor Building. The non-safety-related power sources are backed up by the safety-related EDG. Fires in the Reactor Building would not affect the normal shutdown systems in the steam tunnel or Turbine Building either. The grouping of fire areas of like divisions is coordinated from building to building so that divisional fire areas line up adjacent of each other at the interface between the Reactor and Control Buildings.

Total Fire Damage

Based on the layout and design of the plant, the specific location of a fire within a fire area, fire growth rate and intensity are unimportant as long as the integrated intensity of the fire remains within the capability of the 3-hour fire barrier system. There is no unacceptable failure that can occur due to fire induced failures within a division. Complete burnout of any fire area without recovery will not prevent safe shutdown of the plant. In other words, safe shutdown of the plant can be achieved even if all equipment in any one fire area is damaged by fire and that reentry to the fire area for repairs and operator actions is not possible. Independence of the redundant systems available for shutdown will not be compromised by fire, the consequences of fire, or the failure of fire protection equipment or systems.

Shutdown Capability Independent from MCR

The control room fire area is separated from all other fire areas and the safe shutdown function is backed up by the remote shutdown panel. In the event of loss of the main control room due to fires, the redundant control of the safe shutdown function from outside of the control room is provided by the remote shutdown panel. The remote shutdown system is physically and electrically independent from the control room. A fire resulting in the loss of the control room function will not affect the operation of the remote shutdown system or its support systems. There are no power supply interactions between the control room and the remote shutdown panel because the signals between the remote shutdown panel and the control room are multiplexed over fiber-optic cables.

Inadvertent Operation of Fire Protection System

The Lungmen NPS has design features that reduce the likelihood of inadvertent actuation of fire protection systems and limit the effects of water spray onto safety-related equipment. For example, the EDG room fire detection system is seismically qualified to protect against inadvertent actuation. Automatic foam sprinkling systems are provided for the EDG rooms and EDG day tank rooms. The automatic sprinkler systems used in the EDG rooms are designed to prevent the inadvertent actuation using preaction automatic sprinklers. Two actuation signals are required to initiate the fire suppression system. The first actuation signal will annunciate an alarm locally and in the MCR to alert the operator to any potential problems. Actuation of the sprinklers requires the detection of a fire by infrared and/or rate of heat detectors, and the opening of the fusible link sprinklers. The sunken floor around the EDG can accommodate the contents of the day tank plus 10 minutes worth of foam applied by the automatic sprinkler. This helps to minimize the impact of fire suppression system operation on the diesel operation.

In addition, to protect safety-related equipment from being damaged by the inadvertent discharge of a sprinkler system, adequate drainage is generally provided. For example, any water introduced into the subfloor area of the MCR can be removed by floor drains in the subfloor area or through the use of temporary portable sump pumps. The accumulation and/or drainage of fire water will not affect the ability to safely shut down the reactor. Even with no drainage, the subfloor area in the MCR will accommodate a

minimum of 1 hour's accumulation of water from two hand ROC 50 mm hoses at 130 L/min each without affecting safety-related equipment. If the fire water is assumed to transport immediately to the basement of the Control Building, the resulting accumulation of water will not affect safety-related equipment located in the basement. Accumulation of water in this area will be limited in depth and will not adversely affect water sensitive safety-related equipment located above the floor.

Control and electrical equipment is contained in enclosures, which provide some protection from water spray. Additionally, the raceway design includes features to prevent the ingress of water to a cabinet or panel. Because of the use of fiber optic cables in the multiplexing system for control signals, effects of some of the possible electrical shorting can be prevented. Furthermore, due to the design and layout of the Lungmen NPS, failure of a fire suppression system can impact at most one division of the safe shutdown systems. Therefore, inadvertent operation of the fire protection systems or the occurrence of a single failure in any of these systems will not prevent plant safe shutdown.

Shared Area between Adjacent Units

The swing DG at the Lungmen NPS will be designed to preclude the possibility of common mode failure disabling more than one DG during a fire in the Auxiliary Fuel Building. The shared hot machine shop, radwaste building, service water pump house, circulating water pump house, and related tunnels will be similarly designed to preclude the possibility of a fire-induced common cause failure causing a simultaneous trip demand or impacting safety equipment on both units. A question was asked regarding the fire propagation from the main control room for one unit to the control room for the adjacent unit. No response has been provided by TPC.

19.2.2.2.3.2 Basis for Acceptability

In terms of fire protection, the major improvements in the Lungmen NPS design and fire protection program over the currently operating BWRs include three redundant and independent safety divisions separated by 3-hour fire barriers, use of fiber optic cables in the multiplexing system with dual channels, elimination of the cable spreading room,

seismic design of the FPS piping, fire detection system, and fire barrier systems that may affect safety-related equipment, smoke removal capability, etc.

Separation of the three safety divisions ensures that two redundant safety divisions are free from fire damage and available for safe shutdown. This minimizes the impact of fire damage and effects of the fire suppressant due to either fire fighting efforts or inadvertent operations. The 3-hour fire barrier system ensures that smoke, hot gases, and the fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe shutdown capabilities. Separation of the three safety divisions also ensures that safe shutdown can be achieved even if all equipment in any one fire area is rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible.

The use of fiber optic cables in the multiplexing system considerably lowers the amount of fire sources in certain areas, eliminates cable spreading room, excludes control system interactions between MCR and the remote shutdown panel, and minimizes spurious control actions due to hot shorts. In currently operating plants, cable spreading room is one of the most risk-significant fire areas. The elimination of this room significantly reduces the fire risk. Similarly, the likelihood of control room fire is also decreased due to the reduction of control cables between MCR cabinets and consoles. With no control interactions between the MCR and remote shutdown panel, an independent alternative shutdown capability that is physically and electrically independent of the control room is assured.

Seismic design of certain segments of the fire protection systems also reduces seismic-fire interactions. The smoke removal capability not only improves the effectiveness of manual fire fighting, but also minimizes the likelihood of leakage of smoke, hot gases, combustion products, and fire suppressants from the area experiencing the fire.

Based on the core damage frequency presented in Attachment AD of the Lungmen PSAR Appendix A, the total contribution from fires to the frequency of severe accidents is relatively small compared to other accident initiating events. In addition, no fire areas contribute to more than about 25% of the total fire-induced core damage risk. Therefore, no fire vulnerabilities have been identified for the Lungmen NPS design and fire protection program.

Based on the evaluation above and the discussions given in the preceding section for the fire prevention/mitigation inside the containment, it is concluded that the Lungmen NPS design and fire protection program meet the criteria identified in SECY-90-016 and is acceptable for preventing and mitigating threats from fires for DBAs and severe accidents.

19.2.2.2.4 Intersystem Loss of Coolant Accident

Intersystem loss-of-coolant accidents are defined as a class of LOCAs in which the RCS pressure boundary is breached and coolant is lost through an interfacing system with a lower design pressure. The breach may occur in portions of piping located outside the primary containment, causing a direct and potentially unisolable discharge from the RCS to the environment. An ISLOCA is of concern because of potential direct releases to the environment, loss of core cooling, and loss of core makeup.

High or low pressure interfaces occur on many lines including low-pressure injection lines and the RHR heat exchangers. An ISLOCA occurs when high pressure is introduced in a low-pressure system because of valve failure or an inadvertent valve actuation. In either case, the overpressurization can cause the low-pressure system or components to fail. An ISLOCA concurrent with a loss of all core cooling may lead to core damage.

To protect against ISLOCAs, it was recommended in SECY-90-016 that evolutionary LWR designs reduce the possibility of a LOCA outside containment by designing (to the extent practicable) all systems and subsystems connected to the RCS to an ultimate rupture strength (URS) at least equal to the full RCS pressure. Enhancements of isolation capability or the number of intersystem barriers (e.g., three isolation valves) are not considered to be adequate alternatives in systems that can be practically designed to the URS criteria.

The "extent practicable" phrase shows a realization that all systems must eventually interface with atmospheric pressure and that for certain large tanks and heat exchangers, it would be difficult or prohibitively expensive to design such systems to a URS equal to full RCS pressure. For all interfacing systems and components, which do not meet the full URS criteria, engineering feasibility analysis must be provided as

justification. It was further recommended that systems that have not been designed to withstand full RCS pressure should include (1) the capability for leak testing of the pressure isolation valves, (2) indication in the control room of valve position when isolation valve operators are deenergized, and (3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed.

As indicated in the June 26, 1990 SRM issued by the USNRC, all elements of the low-pressure system must be considered (e.g., instrument lines, pump seals, heat exchanger tubes, and valve bonnets) in the implementation of the above guidance for the protection of ISLOCA.

It was concluded by USNRC that designing, to the extent practicable, low-pressure systems to withstand full RCS pressure is an acceptable means for resolving the ISLOCA issue.

19.2.2.2.4.1 Preventive and/or Mitigative Features

To reduce pressure challenge and resolve the ISLOCA issue, the Lungmen NPS low-pressure piping systems that interface with the reactor coolant pressure boundary (RCPB) will be designed to withstand reactor pressure to the extent practicable. This would minimize the pressure loading experienced by each system and subsystem connected to the RCS should an ISLOCA occur. The URS design pressure for these systems is 2.82 MPaG (i.e., 0.4 times the normal operating RCPB pressure of 7.07 MPaG). This minimum pressure will ensure reasonable protection against burst failure should the low-pressure system be subjected to full RCS pressure. This design guideline applies to all the piping and non-piping components within the boundary for the URS design pressure, including those along any branch line such as RCS drain and vent lines. The non-piping components include flanges, connectors, and packing, including valve stem seals, pump seals, valve bonnets, etc.

The minimum wall thickness of the low-pressure piping designed for URS will be no less than that of a standard weight pipe. Valves will be designed to Class 300 as a minimum (a Class 300 valve is adequate for ensuring the pressure of the low-pressure piping system under full reactor pressure). The design of the low-pressure piping systems will

be in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subarticle NC/ND-3600. Periodic surveillance and leak rate testing of the low pressure system isolation valves will be required in accordance with the Technical Specification requirements as a part of the ISI program.

For some low-pressure systems attached to the RCPB, it is not practical or necessary to provide a higher system ultimate pressure capability for the entire low-pressure connected system. This applies to large tanks that are vented to the atmosphere and have a low design pressure. Also included as impractical to design to the URS design pressure is the suppression pool that provides a low-pressure sink for its interfacing systems. If the RCPB interfacing piping is not designed to the URS pressure, the piping should be designed to provide (1) the capability for leak testing of the pressure isolation valves, (2) valve position indication that is available in the control room when isolation valve operators are deenergized, and (3) high-pressure alarms to warn control room operators when rising reactor pressure approaches the design pressure of attached low-pressure systems or when both isolation valves are not closed. However, based on the TPC's response to a question raised during the review, it appears that these three alternative protection features will not be provided to any of the low pressure systems that interface with the RCPB and are not designed to the URS. This remains an open item.

Fourteen Lungmen NPS systems, which have been determined to interface directly or indirectly with the RCPB, have been evaluated to ensure resistance to ISLOCA by a minimum design pressure of 2.82 MPaG. The low pressure piping boundaries for this minimum design pressure are extended to the last closed valve connected to piping interfacing a low pressure sink. The Lungmen NPS heat exchangers are not affected by ISLOCA and will not be designed to the URS design pressure. These include RWCU, RHR, and FPCU heat exchangers.

The periodic surveillance testing of the ECCS injection valves that interface with the reactor coolant system might lead to ISLOCA conditions if their associated testable check valves was stuck open. To avoid this occurrence, the RHR, HPCF, and RCIC motor operated injection valves will only be tested during low pressure shutdown conditions. The RHR shutdown cooling suction line containment isolation valves will also only be tested during shutdown operation. These valves are interlocked against

opening for reactor pressure greater than the shutdown cooling setpoint of approximately 0.93 MPaG.

19.2.2.2.4.2 Basis for Acceptability

It was concluded in NUREG-1503 that using a design pressure equal to 0.4 times the normal RCS operating pressure of 7.07 MPaG and using a minimum wall thickness no less than a standard weight pipe can provide an adequate basis for assuring that the low pressure piping systems that interface with the RCS pressure boundary can withstand full reactor pressure and meet the recommendations in SECY-90-016 and its applicable regulation for designing against ISLOCAs. Therefore, the Lungmen NPS piping systems with a URS design pressure of 2.82 MPaG will provide an ultimate rupture strength equal to or greater than the normal operating RCPB pressure of 7.07 MPaG, and will ensure reasonable protection against burst failure should the low pressure systems be subjected to full RCS pressure. A systematic evaluation of interfacing systems was performed for the low-pressure systems at the Lungmen NPS to ensure the URS protection and it was determined that the SECY-90-016 requirements were satisfied.

In conjunction with the additional design guidelines described in the preceding section for the Lungmen NPS, it is judged that

1. The likelihood of the low-pressure piping rupturing under full RCS pressure is low,

2. The likelihood of intolerable leakage is low under ISLOCA conditions although some leakage may occur at flanges and valve bonnets, and

3. Some piping components might undergo gross yielding and permanent deformation under ISLOCA conditions.

As such, the frequency of severe accident resulting from ISLOCA at the Lungmen NPS is expected to be much lower than that for the currently operating BWRs due to the higher design pressure in the low pressure piping and components that may interface with the reactor coolant pressure boundary.

Based on the above evaluation and discussions presented in the preceding section, it is concluded that the Lungmen NPS has met the criteria defined in SECY-90-16 regarding ISLOCA prevention and mitigation for the advanced reactors. There is reasonable assurance that the low-pressure piping systems interfacing with the RCPB are structurally capable of withstanding the consequences of an intersystem loss-of-coolant accident. To ensure that there are no vulnerabilities in the Lungmen NPS design to prevent and mitigate the ISLOCA event, TPC has committed to quantitatively evaluate the ISLOCA (and LOCA outside containment) in the FSAR Appendix A.

19.2.2.3 Site Specific External Hazards: Earthquakes And Typhoons

Earthquake and typhoon are the two most significant external threats to the nuclear plants in Taiwan. The most important impact of typhoons on the Lungmen NPS is the potential loss of offsite power. Earthquakes can cause damage to plant equipment and buildings in addition to loss of offsite power. Currently, the core damage frequency estimated for the Lungmen NPS is completely dominated by the seismic events. These external hazards were evaluated in Appendix A to the Lungmen PSAR, but not discussed in Chapter 19. To address severe accident concerns, consideration of the dominant contributors to the severe accident risk and the insights identified by the PRA is necessary for the assurance of no undue risk to public health and safety. TPC indicates that external hazards have been evaluated deterministically in Chapter 3 and probabilistically in Appendix A. However, a summary from the perspective of severe accident prevention should be included in Chapter 19. No commitment has been made by TPC to include this in Chapter 19. It remains an open item.

19.2.2.4 Impact of Differences between Lungmen NPS and Standard ABWR

The effects of differences in design and operations between the Lungmen NPS and the Standard ABWR on severe accident prevention are not addressed in Chapter 19 of the Lungmen PSAR. TPC has committed to summarize in the FSAR Chapter 19 differences in design and operations between the final Lungmen NPS design and Standard ABWR that are relevant to severe accident prevention and mitigation. However, no commitment has been made by TPC to provide a qualitative discussion of the impact of each of these differences on the various aspects of severe accident prevention. This remains an open item.

19.2.3 Summary of Assessment

Based on the discussions presented in the preceding sections and the low estimated core damage frequency, it is concluded that, with the exception of some open items, the preliminary Lungmen NPS design has been shown to be acceptable for severe accident prevention. It has demonstrated compliance with the requirements of the applicable regulations and the criteria set forth for this evaluation.

19.3 Deterministic Assessment of Severe Accident Mitigation

19.3.1 Overview of the Containment Design

The Lungmen containment structure is a reinforced concrete right circular cylinder with a steel upper drywell head. The cylindrical containment wall is 2 m thick and has an inside radius of 14.5 m. The structure has a height of 29.5 m measured from the top of the foundation mat to the bottom of the containment top slab. The internal surface of the Lungmen containment is lined with steel plate. The floor of the lower drywell includes a 1.5 m layer of concrete above the containment liner to ensure that debris will not come in direct contact with the containment boundary upon discharge from the reactor vessel. The Lungmen containment is supported by a foundation mat that is 5.5 m thick. The bottom of the foundation mat is embedded in the ground approximately 26 m below grade.

The containment top slab is integral with the fuel pool girders and the containment wall. The top slab is nominally 2.2 m thick but is 2.4 m thick beneath the fuel pool, and the steam dryer and separator pool.

The containment interior is divided by the diaphragm floor and the reactor pedestal into an upper drywell region, a lower drywell region, and a wetwell region. The reactor pedestal is a composite steel and concrete structure that provides support for the reactor pressure vessel, the reactor shield wall, the diaphragm floor, access tunnels, horizontal vents, and the lower drywell access platforms. The reactor pedestal consists of two concentric steel shells tied together by vertical steel stiffener plates and filled with concrete. Ten downcomer vents are built into the reactor pedestal and connect the

upper and lower drywell regions. These vents extend downward through the pedestal and each steel vent line connects to horizontal vent outlets into the suppression pool.

The diaphragm floor is a reinforced concrete circular slab with an outside radius of 14.5 m and a thickness of 1.2 m. The diaphragm floor serves as a barrier between the drywell and the wetwell.

There are two 4.3 m diameter access tunnels that penetrate the containment suppression pool wall and the reactor pedestal. The centerlines of these tunnels are located approximately 8.0 m above the top of the base slab.

The wetwell region consists of a suppression pool and air volume. Steam from a blowdown of the reactor vessel or from a pipe break inside the drywell is passed through the SRVs or downcomer vents and condensed in the suppression pool. A system of vacuum breakers connecting the drywell and wetwell regions is provided to prevent excessive differential pressures.

The containment atmosphere is maintained at inert conditions by an Atmospheric Control System (ACS). The wetwell-to-drywell vacuum breaker system is a subsystem of the ACS. In addition, the ACS also (1) assures that primary containment integrity is maintained by providing primary containment isolation, and (2) includes a subsystem known as the Containment Overpressurization Protection System (COPS) that protects the containment from overpressure failure by venting steam and non-condensable gases from the wetwell air space through the plant stack. The COPS vent lines contain two rupture disks in series to relieve containment pressure. The pressure setpoint of the disk exposed to the wetwell atmosphere (i.e., the inner disk) is established at a level above the containment design pressure but below the ASME Service Level C capability of the containment. The setpoint of the outer rupture disk, located nearest the plant stack, is relatively low.

The containment heat removal function is accomplished with the Containment Cooling System (CCS) which encompasses several of the RHR operating modes. These modes include the Low Pressure Flooder mode, the Suppression Pool Cooling mode, and the Containment Spray modes (drywell and wetwell). A capability exists to connect the drywell spray header to the fire protection system pumps, one of which is driven by a diesel engine.

A capability to flood the lower drywell is provided by the Lower Drywell Flooder System (LDF). This system provides a path for suppression pool water to flow into the lower drywell region during a severe accident that leads to core meltdown, vessel failure, and the release of molten core debris to the lower drywell floor. The LDF consists of ten pipes that run from the vertical pedestal vents to the lower drywell. Each of these pipes contains a fusible plug valve connected to the end of the pipe that extends into the lower drywell. The fusible plug valves open when the lower drywell atmosphere reaches a temperature of approximately 260 °C.

19.3.2 Severe Accident Progression

This section contains a description of the processes that may occur during the progression of a severe accident and how these phenomena affect containment performance. Severe accident performance can be divided into two phases: an in-vessel phase and an ex-vessel phase. The in-vessel phase generally begins with inadequate decay heat removal and can lead to melt-through of the reactor vessel. The ex-vessel phase begins with the release of core debris from the reactor vessel into the containment and includes phenomena such as core-concrete interactions (CCI), fuel-coolant interactions (FCI), and direct containment heating (DCH) resulting from high pressure melt ejection (HPME) of the debris.

19.3.2.1 In-Vessel Melt Progression

For severe accidents that proceed to vessel failure and the release of molten core material into the containment, the in-vessel phase of melt progression establishes the initial conditions for assessment of the thermal and mechanical loads that potentially threaten the integrity of the containment. The in-vessel melt phenomena start with uncovering of the core and initial heat-up, and continue until either (1) the degraded core is stabilized and cooled within the reactor vessel, or (2) the reactor vessel is breached and molten core material is released into the containment. The phenomena and processes that can occur during the in-vessel phase of melt progression and are applicable to the Lungmen NPS include the following:

- Core heat-up due to inadequate cooling.

- Clad oxidation and metal-water reaction.
- Formation of eutectics as a result of interactions between core materials (e.g., control blades and fuel assembly channel boxes). Eutectics are mixtures of materials with a melting point lower than that of any other combination of the same components.
- Melting and relocation of clad, structural materials, and fuel.
- Drainage of molten materials to the vessel lower head region (dry-core scenarios).
- Formation of blockages near the bottom of the core resulting from the solidification of relocating molten materials (wet-core scenarios).
- Formation of a melt pool, natural circulation heat transfer, crust formation, and crust failure (wet-core scenarios).
- Breach of the reactor vessel lower head due to failure of a penetration or local or global creep-rupture.

Adequate core cooling can be defined as providing sufficient cooling water flow to the reactor core to remove the decay heat produced. If the decay heat is being transferred to the containment, adequate core cooling is defined as providing sufficient cooling water to containment heat removal systems to remove the decay heat transferred from the core. The mechanisms by which sufficient cooling is provided to the Lungmen reactor core or the containment are numerous, diverse, and redundant. These mechanisms include both safety and non-safety systems. Examples of safety systems include the RCIC system, high-pressure core flooder systems, RHR system, and ADS. Examples of non-safety systems include the condensate and feedwater systems, main condenser, turbine bypass valves, and the ACIWA system.

In the event that all safety and non-safety systems fail to remove the decay heat being produced, the core will heat up to the point where damage to fuel and fuel clad may occur. Decay heat is transferred through radiative, conductive, and convective heat transfer to the steam, other core materials, and non-fuel materials within the reactor. Inadequate cooling results in coolant boiloff and a decreasing level within the reactor

vessel as the decay heat generation exceeds the heat removal rate. If adequate cooling cannot be restored, the coolant level within the core continues to decrease to levels below the top of the active fuel. That portion of the fuel rods above the coolant level is cooled only by rising steam. The uncovered fuel rods begin to overheat and Zircaloy clad oxidation begins in the presence of steam at high temperatures. As the clad oxidizes in the presence of steam, hydrogen and additional heat are generated.

As the fuel rods continue to heat up from decay heat and the exothermic zirconium oxidation reaction, the materials within the reactor with low melting points are expected to melt first and may form eutectics.

Zircaloy has a melting point of approximately 1,760 °C and when it begins to melt, the protective layer of ZrO_2 breaks down, exposing unoxidized zircaloy. Local melting of the fuel rods can cause changes in the core geometry resulting in different steam flow paths. These changes can potentially affect the oxidation process in different ways. If additional access to unoxidized zircaloy is created by the changes, additional oxidation will occur. In contrast, if the melt formation or changes in the steam flow path reduce the zircaloy surface available for oxidation, the extent of oxidation will decrease the overall reaction process. In some accident scenarios in which residual amounts of water remain in the bottom of the core and lower plenum, substantial steaming and oxidation can take place.

In addition to oxidation, the potential exists for zircaloy to interact with the UO_2 fuel, forming eutectics. The formation of eutectics may decrease the effective surface area for oxidation and the overall oxidation rate. The melting point of zircaloy is dependent upon its oxidation state and lattice structure. It has three melting points that include 1,877 °C (beta-Zr), 1,977 °C (alpha-Zr(O)), and 2,677 °C (ZrO_2). When partially oxidized zircaloy is in contact with UO2, an alpha-Zr(O)/UO_2 based eutectic will form with a liquefaction temperature of approximately 1,900 °C . Therefore, in the presence of good fuel or cladding contact, fuel liquefaction and melt relocation will commence at approximately this temperature.

The results of various severe-fuel damage test programs sponsored by the USNRC (e.g., those discussed in References 19-10 and 19-11) indicate that oxidation of zircaloy is largely controlled by the availability of a steam supply and that high rates of hydrogen

generation can continue after melt formation and relocation. Some of the USNRC sponsored experiments indicate that the majority of the hydrogen generated occurs after onset of zircaloy melting and fuel dissolution. In steam-rich experiments, oxidation took place over most of the fuel bundle length, and most of the hydrogen is generated early. For steam-starved experiments, oxidation was limited to local regions of the fuel bundle and the majority of the hydrogen was generated after the onset of Zr/UO_2 liquefaction and relocation.

Hydrogen production and accumulation could present significant challenges to the containment, including deflagration, detonation (or deflagration-to-detonation transitions), and overpressurization. The Lungmen NPS containment atmosphere will be made inert with nitrogen to prevent the occurrence of any deflagration or detonation. Due to the COPS feature, pressurization of the containment from the generation and accumulation of hydrogen and other gases will not exceed ASME Service Level C limits.

The Flammability Control System (FCS) will be manually operated if hydrogen is present. In response to a question regarding the possibility of an operator induced deinertion and/or ignition, TPC noted that deinerting will be controlled by technical specifications and plant procedures. Deinerting will be allowed for no more than 24 hours before (shutdown) or after (startup) 15 % power is reached. The FCS is designed to recombine hydrogen and oxygen generated by metal-water reactions during LOCA. Although manually initiated, the LOCA initiation signal from the main control room is used to start the system. TPC notes that the chance of an operator error that will deinert the containment especially with hydrogen present in excess of the flammability limit is extremely remote.

Severe fuel damage tests have indicated the potential for incoherent melt relocation due to noncoherent temperatures within the test bundles. This is because of the different core materials present with a wide range of melting points and eutectic temperatures. Formation of eutectics would result in a nonuniform melting and relocation process. Additional variations in the melt-relocation process can be attributed to asymmetric bundle heating that can increase because of zircaloy oxidation. This process begins when one area of the fuel bundle is initially at a temperature higher than the other areas. The higher-temperature zircaloy will consume the available steam through oxidation at a quicker rate.

As the temperature of the core increases, fission products in vapor form are released. The mixing and transport of these fission products within the primary system depend upon flow paths set up by any existing steam or hydrogen and any interactions with surfaces within the reactor. Such surfaces as the upper internal structures of the reactor vessel may act as a filter where microscopic fission product aerosols suspended in the gas can settle on comparatively cool surfaces by thermophorisis and diffusiophoresis. In addition, retention mechanisms such as turbulent deposition and gravitational settling occur.

The progression of core melting, including relocation and fission product release, becomes increasingly difficult to predict as the degradation process proceeds. The molten core material could relocate into the lower plenum of the reactor vessel, and if water is present in this plenum, the potential exists for in-vessel steam explosions. Steam explosions occur when molten fuel rapidly fragments into small particles and transfers its energy to the water, causing rapid steam generation and shock waves. Another possibility is that the core debris within the lower plenum may quickly melt through the reactor vessel or interact with available water before melting through and entering the lower drywell.

As indicated, the in-vessel accident progression, including core degradation, relocation, and failure of the reactor vessel, is rather uncertain. This uncertainty results from various phenomena, including the potential for in-vessel steam explosion, the interaction of core debris with internal vessel structures, the time and mode of vessel failure, the composition of the core debris released at vessel failure, the amount of in-vessel hydrogen generation, the in-vessel fission product release, and the transport and retention of fission products and other core materials in the RCS.

19.3.2.2 Ex-Vessel Melt Progression

Ex-vessel severe accident progression is affected by the mode and timing of the reactor vessel failure; the primary system pressure at the time of reactor vessel failure; the composition, amount, and character of the molten core debris that is expelled from the reactor vessel; the type of concrete used in the containment structures that could come in direct contact with core debris; and the availability of water to the lower drywell region. The initial response of the containment to ex-vessel severe accident progression is

largely a function of the pressure of the RCS at the time of reactor vessel failure and the presence of water within the reactor cavity prior to vessel breach. If not precluded by design features, consequences to the public are usually dominated by early containment failure mechanisms that could result from energetic severe accident phenomena such as HPME /DCH and ex-vessel steam explosions. The long-term response of the containment from ex-vessel severe accident progression is largely a function of the containment pressure and temperature resulting from core-concrete interaction and the availability of containment heat removal mechanisms.

At high RCS pressures, the molten core debris could be ejected from the reactor vessel in the form of a jet, causing it to fragment into small particles. The potential exists for the core debris ejected from the vessel to be swept out of the lower drywell and into the upper drywell. Finely fragmented and dispersed core debris could heat the containment atmosphere and lead to large pressure spikes. In addition, chemical reactions between core debris particulates and oxygen and steam could add to the pressurization loads. This severe accident phenomenon is known as HPME with DCH.

To prevent HPME, the Lungmen NPS has incorporated a reliable depressurization system to provide assurance that in the event of a core melt scenario, failure of the RPV would occur at low pressure. Should the RPV fail at a high pressure, the design of the Lungmen NPS containment provides only an indirect pathway from the lower to the upper drywell, decreasing the amount of core debris that could contribute to DCH.

It was not clear during the staff's review as to whether the equipment tunnels provided a pathway for debris dispersal during HPME. TPC was asked to clarify if such dispersal could impact the progression of severe accidents. TPC's response to questions raised in this regard states that debris accumulation is not an issue since the elevation of the tunnels is much higher than the largest possible debris depth. The TPC response does not address the potential for debris accumulation in the tunnels due to HPME, the potential for melt-through, and the possibility of suppression pool bypass. This remains an open item.

If water is present in the lower drywell at the time of RPV failure, the resulting FCI has the potential for rapid steam generation or steam explosions. Rapid steam generation involves the pressurization of containment compartments from nonexplosive steam

generation beyond the capability of the compartment to relieve the pressure so that local overpressurization failure of the compartment occurs. Steam explosions involve the rapid mixing of finely fragmented core debris with surrounding water resulting in rapid vaporization and acceleration of surrounding water creating substantial pressure and impact loads. The Lungmen NPS is designed so that there is a very low likelihood of water within the lower drywell at the time of reactor vessel failure.

The eventual contact of molten core debris with concrete in the lower drywell will lead to core-concrete interaction. CCI involves the decomposition of concrete from core debris and can challenge the containment in various ways, including (1) pressurization resulting from the production of steam and noncondensible gases, (2) the transport of high temperature gases and aerosols into the upper drywell leading to high-temperature induced failure of containment seals and penetrations, (3) liner melt-through, (4) reactor pedestal melt-through leading to relocation of the reactor vessel and tearing of containment penetrations, and (5) the production of combustible gases such as hydrogen and carbon monoxide. CCI is affected by many factors including the availability of water to the lower drywell, the geometry of the containment, the composition and amount of core debris, the core debris superheat, and the type of concrete involved.

The Lungmen NPS has incorporated a number of design features to mitigate the effects of CCI. These features include an LDF system, an ACIWA system, a sacrificial layer of basaltic concrete for the lower drywell floor, and the COPS. The LDF system is passively actuated and provides suppression pool water to assist in cooling core debris once it has entered the lower drywell. The ACIWA system provides for both reactor vessel injection and drywell spray capability to cool core debris or control containment pressurization. Basaltic concrete protects the containment liner from melt-through and decreases the amount of noncondensible gases generated during CCI when compared with limestone-based concretes. The COPS is designed to passively relieve containment pressure to prevent gross containment failure during severe accidents when the containment pressure approaches ASME Service Level C limits. The COPS relief pathway takes advantage of the scrubbing capability of the suppression pool to limit any offsite releases.

19.3.3 Severe Accident Mitigative Measures

19.3.3.1 Hydrogen Generation and Control

The generation and potential combustion of large quantities of hydrogen in a severe accident could pose a threat to containment integrity. The major source of hydrogen generation during the in-vessel phase of a severe accident is the oxidation of core zirconium by steam when the zirconium achieves temperatures well above normal operating levels. The degree of oxidation is dependent on a variety of parameters related to severe accident event progression. These parameters include RCS pressure, the timing and flow rate of reflooding if it occurs, and the reactor core temperature profile.

The generation of hydrogen as a result of ex-vessel core debris reacting with steam or concrete must also be considered.

19.3.3.1.1 Preventive and/or Mitigative Features

10 CFR 50.34(f) requires a system for hydrogen control that can provide with reasonable assurance that uniformly distributed hydrogen concentrations in the containment will not exceed 10 percent during and following an accident that releases an amount of hydrogen equivalent to a 100 percent fuel-clad metal water reaction, or that the post accident atmosphere will not support hydrogen combustion. To comply with 10 CFR 50.34(f), the Lungmen containment atmosphere will be inerted during normal operation to prevent hydrogen deflagrations/detonations during a severe accident.

10 CFR 50.34(f) also requires that containment integrity be maintained below ASME Service level C limits for steel containments and the factored load category for concrete containments during an accident that releases hydrogen generated from 100 percent fuel-clad metal water reaction. TPC summarized an analysis in Section 19.4.3.1.2 of the PSAR that assessed the capability of the Lungmen containment to withstand pressurization resulting from 100 percent fuel-clad metal water reaction coupled with the

effects of a large-break LOCA. This analysis resulted in a peak containment pressure of approximately 0.618 MPa absolute.1

As noted earlier, the Lungmen NPS has a concrete containment with a steel upper drywell head. The governing Service level C (for steel portions not backed by concrete)/factored load category (for concrete portions including the steel liner) pressure capability goal of the containment structure is 0.77 MPa absolute. This pressure capability goal corresponds to the internal pressure required to cause the maximum stress intensity in the steel drywell head to reach general membrane yielding according to Service Level C limits of ASME Section III, Division 1, Subarticle NE-3220. Therefore, the Lungmen containment has the capability to withstand 100 percent fuel-clad metal-water reactions as required by 10 CFR 50.34(f). The ability to cool the debris ex-vessel is necessary to prevent corium-concrete interaction (CCI) and/or thermal attack of the molten corium on containment structures.

In its analysis of hydrogen generation, TPC found that even with the disabling of the blockage and eutectic cutoff models, MAAP-ABWR predictions of the extent of clad oxidation did not exceed approximately 36 % for realistic scenarios. Non-mechanistic scenarios and changes in model parameters were required to produce 100 % clad oxidation.

19.3.3.1.2 Basis for Acceptability

The Lungmen containment design meets the criteria of SECY-90-016 (Reference 19-2) and 10 CFR 50.34(f) by utilizing a nitrogen inerted atmosphere within the containment. The Lungmen containment is cable of withstanding the combined pressure loadings resulting from a large-break LOCA and hydrogen generation equivalent a 100 percent fuel-clad metal water reaction as required by 10 CFR 50.34(f).

---

1 It should be noted that in the version of the Lungmen PSAR that was reviewed, not all of the references to pressure were specifically cited as being in terms of absolute pressure or gauge pressure. TPC has committed to adopt the practice utilized during the US ABWR Certification Project in future revisions.

19.3.3.2 Core Debris Coolability

Corium-concrete interaction is a concern for severe accidents that involve a release of molten corium to the drywell floor. If the molten debris can not be adequately cooled, the transfer of heat to the drywell floor and surrounding structures will lead to decomposition of the concrete, the generation of non-condensable gases and aerosols, possible structure damage, and possible failure of the containment liner. All of these mechanisms result in a challenge to or failure of containment integrity.

Coolability and quenchability have been the subject of extensive research for a number of years. However, much uncertainty still exists regarding these phenomena and these uncertainties will most likely not be resolved in the near future. Because of this uncertainty, the USNRC certification review of the Standard ABWR did not address the question of whether coolability or quenchability can be achieved, but rather, what the impact on the containment design is if they are not achieved. This same philosophy has been adopted for this review.

As noted above, CCI involves the melting and decomposition of concrete that comes into contact with molten corium. This phenomenon could occur during accident sequences that result in the release of molten corium to the lower drywell floor following a breach in the reactor vessel. The thickness of the corium layer within the lower drywell depends upon the amount of core debris released from the reactor vessel, its spreadability, and the lower drywell floor area. Once on the drywell floor, the molten corium may react with the concrete and any available water, producing noncondensible gases, water vapor, and heat from exothermic reactions if metallics are present in the debris.

CCI can challenge the containment function in various ways including pressurization due to the generation of noncondensible gases and steam, destruction of structural support members, and melt-through of the containment liner. Noncondensible gases, primarily carbon dioxide, carbon monoxide, and hydrogen, are released from the concrete as it decomposes and are formed from reactions between water and metals within the molten corium. The corium and concrete are heated from the combined effects of decay heat and exothermic chemical reactions.

19.3.3.2.1 Preventive and/or Mitigative Features

It was recommended in SECY-93-087 (Reference 19-4) that evolutionary and passive light water reactor designs: (1) provide reactor cavity floor space to enhance debris spreading, (2) provide a means to flood the reactor cavity to assist in the cooling process, (3) protect the containment liner and other structural members with concrete, if necessary, and (4) ensure that the best estimate environmental conditions (pressure and temperature) resulting from CCI do not exceed Service Level C for steel containments or factored load category for concrete containments for approximately 24 hours.

The Lungmen containment incorporates numerous features to mitigate the effects of CCI. The following features were judged to be the most important: a large drywell floor area and shields to prevent corium from flowing into the sumps, a Lower Drywell Flooder System, an AC-Independent Water Addition System, a layer of sacrificial basaltic concrete above the liner in the drywell floor, a thick walled reactor pedestal, and a Containment Overpressure Protection System.

19.3.3.2.1.1 Lower Drywell Floor Area

The lower drywell floor of the Lungmen containment has minimal obstructions and an area (not including the cross-sectional area of the sumps) of 79 m^2. This floor area is sufficient to satisfy the EPRI design criteria of 0.02 m^2/MWt for debris coolability.

The lower drywell floor contains two sumps; an equipment drain sump and a floor drain sump that could result in the accumulation of core debris and accelerated CCI. However, debris flow into these sumps is prevented by corium shields that extend from the floor of the sump to a height above the drywell floor that is greater than the expected maximum height of the core debris bed. The shield consists of a protective layer of refractory bricks (alumina).

It is concluded that the unobstructed floor area of the lower drywell, along with the design features mentioned above, provides measures to promote the potential for core debris coolability, but does not necessarily ensure it. No endorsement of the EPRI floor sizing criterion should be implied.

19.3.3.2.1.2 Lower Drywell Flooder System

The LDF provides an alternate source of water for flooding the lower drywell region in the event of core debris discharge from the reactor vessel and failure of the ACIWA. The LDF consists of ten lines connecting the suppression pool to the lower drywell. Each line contains a thermally activated flooder valve that is actuated by the melting of a fusible plug. The temperature setpoint of the fusible plug is 260 °C. The inlet to the lines is located approximately 4 m below the normal suppression pool water level and the discharge end of the line is approximately 1 m above the drywell floor. Each valve/line is capable of a flow of 10.8 kg/sec. Only two of the valves are required to open to remove decay heat and exothermic heat resulting from the zirconium-water reaction.

Each flooder valve contains four components: a stainless steel disk, a Teflon disk, a fusible metal plug, and a plastic cap. The stainless steel disk prevents suppression pool water from corroding the Teflon disk and fusible metal plug. The Teflon disk provides an insulating barrier to prevent the suppression pool water from contacting the fusible metal plug. This insulating barrier is needed to assure that the fusible metal plug is not cooled by the suppression pool water and prevented from melting. The Teflon disk will not melt or stick in the valve because its softening temperature is approximately 400 °C and its chemical resistance is higher. Accordingly, the Teflon disk will not adhere to the stainless steel plug or to the fusible plug. The fusible metal plug has a small, raised, annular ring around its circumference approximately 2.0 mm high. It is this annular ring that has to melt in order for the LDF to actuate.

The fusible metal plug is made of an alloy mixture of two or more metals so that the plug melts when its temperature reaches 260 °C. The end of the flooder valve line is covered with a plastic cover with a melting point below 130 °C. This plastic cover prevents corrosion of the fusible metal material from the intrusion of moisture. The flooder valves are mounted in the vertical position so that the fusible metal faces downward to facilitate opening of the valve when the melting temperature of the fusible metal is reached. Heat transfer resulting in melting of the fusible plug occurs through a combination of conduction, convection, and radiation. Heat is conducted from the stainless steel pipe to the fusible plug causing it to melt. Heat is also received from the atmosphere within the

lower drywell through convection. In addition, the stainless steel pipe also receives radiative heat from the corium on the lower drywell floor.

During each refueling outage, the 10 fusible plug flanges and outlets will be inspected to ensure there is no leakage. Once every two refueling outages, 2 of the 10 fusible plugs will be removed, inspected, and tested to confirm their function and verify the temperature setpoint.

Based on the above discussion, it is concluded that the LDF meets the criteria of SECY-93-087 for providing a means to flood the reactor cavity to assist in the cooling process of core debris.

19.3.3.2.1.3 AC-Independent Water Addition System

The ACIWA system consists of piping and valves that connects the Lungmen fire protection system to either the Low Pressure Flooder (LPFL) header inside the reactor vessel or to the drywell spray header for the upper drywell.

The primary point of injection for this system is the LPFL header inside the reactor vessel. Operation of the ACIWA in the vessel injection mode is intended to prevent core damage. In the event that the system is not actuated in sufficient time to prevent core damage and vessel melt-through occurs, ACIWA water will flow through the breach to the lower drywell floor at a sufficient capacity to cover the core debris on the floor, provide debris cooling, and scrub fission products released as a result of continued CCI. Depending on the specific accident sequence, ACIWA system induced flooding could preempt LDF induced flooding (ACIWA flooding prevents lower drywell temperatures form reaching the LDF actuation setpoint) or both systems could provide for debris cooling.

Within the fire protection system, an independent diesel-driven pump exists that would provide the pumping capability. Additionally, an external hookup outside the reactor building for connection of a fire truck pump to an alternate water source is provided. Although an electric (ac)-driven fire pump is included within the fire protection system, its contribution to severe accident prevention and mitigation has been excluded because of its dependence on electrical power.

Operation of the ACIWA system in the upper drywell spray mode controls atmospheric temperatures in the upper drywell region and provides fission product scrubbing in addition to the features related to vessel injection. The ACIWA system is capable of injecting water into the containment up to the setpoint of the COPS. Accordingly, the drywell spray mode of operation can significantly delay the time to reach the COPS setpoint or prevent the containment atmosphere from reaching the setpoint at all.

It is concluded that the ACIWA provides a second means (in addition to the LDF) to flood the lower drywell to assist in the process of cooling core debris, as specified in SECY-93-087.

19.3.3.2.1.4 Sacrificial Basaltic Concrete

The lower drywell floor containment floor contains a 1.5 m layer of basaltic concrete above the containment liner. This sacrificial layer is provided to ensure that debris will not come in direct contact with the containment boundary (liner) upon discharge from the reactor vessel. The concrete type selected has approximately 4 weight-percent calcium carbonate, which results in low gas generation rates when subjected to CCI. Reduced noncondensible gas and water vapor production from basaltic concrete has been observed during testing (References 19-12 and 19-13). Lower gas generation rates reduce the rate of containment pressurization, extending the time to potential failure and providing additional time for aerosol removal.

Although basaltic concretes have lower melting points and faster ablation rates than limestone and limestone-common sand concretes, it is concluded that the 1.5 m layer of basaltic concrete meets the criteria in SECY-93-087 relating to protection of the containment liner and provides sufficient protection for the containment liner. No sensitivity studies were performed by TPC relative to the type of concrete used for the sacrificial layer. TPC notes, however, that MAAP-ABWR calculations for the most likely accident scenarios confirm that the sacrificial bed has an adequate depth to protect the containment liner.

Based on the observations discussed above and engineering judgment, it is concluded that the 1.5 m layer of basaltic concrete provides sufficient protection for the containment

liner and meets the criteria specified in SECY-93-087 relating to protecting the containment liner.

19.3.3.2.1.5 Reactor Pressure Vessel Pedestal

The layer of basaltic concrete discussed in the previous section protects the containment liner from CCI in the vertical direction. Thermal attack by the debris in the radial direction could affect the RPV pedestal. The Lungmen RPV pedestal is formed of two concentric steel shells connected by steel web stiffeners. Ten wetwell-to-drywell vents pass through the annular regions between the cylinders. The remainder of the space between the cylinders is filled with concrete. The thickness of the pedestal is 1.7 m.

The inside diameter of the pedestal forms the outer boundary of the lower drywell and is therefore, the barrier to the radial flow of corium. If corium reaches the pedestal, the inner steel cylinder would be attacked by the debris and the pedestal concrete between the cylinders would be subject to ablation. Substantial ablation could lead to failure of the pedestal and subsequent collapse of the reactor pressure vessel and diaphragm floor leading to gross containment failure.

In Section 19.4.3.2.2.3 of the Lungmen PSAR, TPC states that the design of the pedestal is based on the maximum stress obtained in the steel plates. No credit is given to the strength of the concrete. TPC further states that the allowable stress in the steel plates is 0.6 times the yield stress, neglecting temperature. The calculated stress without seismic loads is stated as being 0.4 times the yield strength.

Based on the information provided in Section 19.4.3.2.2.3 of the Lungmen PSAR and calculations performed by the USNRC during its review of the GE Standard ABWR (NUREG-1503), it is concluded that adequate margin exists to the yield strength of the RPV pedestal following 1.5 m of radial ablation and that the RPV pedestal is thick enough to withstand the effects of radial ablation resulting from CCI. These attributes meet the criteria specified in SECY-93-087 for protecting structural members with concrete.

19.3.3.2.1.6 Containment Overpressure Protection System

The COPS is designed to relieve containment pressure before it reaches ASME Service Level C conditions. The COPS provides a pathway for a controlled release that first passes through the suppression pool before being released to the environment from the plant stack. The COPS consists of a series of rupture disks and normally open valves in a line connecting the wetwell air space to the plant stack. The normally open valves can be closed to re-establish containment integrity during post accident recovery. For slow pressurization, the controlled release occurs at a containment pressure of 0.72 MPa absolute (the setpoint of the inboard rupture disk at 93 °C). The outboard rupture disk (located nearest to the stack) is designed to fail at a differential pressure of 0.03 MPa.

The setpoint of the COPS was established based on the competing goals of minimizing the probability of a containment structural failure and maximizing the time to a release of fission products to the environment. TPC assumed that the inboard rupture disk would fail at ± 5 percent of its setpoint at nominal temperature. The effect of temperature on the rupture disk failure pressure was determined to be small, approximately 2 % for a temperature change of 56 °C.

The area of the COPS rupture disk is designed to be sufficiently large to be effective in mitigating the containment pressure increase during an ATWS event in which the operators control the injection flow. The minimum steam flow capacity of the COPS is 28 kg/sec when the containment is at the rupture disk setpoint. TPC notes that this flow capacity provides ample margin to offset the rate of steam generation corresponding to decay heat. TPC also notes that pool swell and blowdown loads resulting from the blowdown of the containment following rupture disk failure will not threaten piping and that significant entrainment will not occur.

The COPS passively relieves containment pressurization before containment pressure reaches ASME Service Level C limits and provides a means for a controlled release through a containment vent pathway with fission product scrubbing provided by the suppression pool. With respect to CCI, the COPS prevents catastrophic overpressurization failure of the containment for severe accident sequences involving prolonged periods of CCI. The COPS ensures that containment pressurization resulting

from CCI does not exceed the ASME Service Level C limit of 0.77 MPa absolute since its actuation setpoint is 0.72 MPa absolute.

Section 19.4.3.2.1.6 of the Lungmen PSAR does not address the capability of the COPS relative to pressure spikes (i.e., sudden pressurization). TPC has committed to adding a discussion in Section 19.4.3.2.1.6 in subsequent revisions to the PSAR that addresses the question of pressure spikes and the COPS design bases.

19.3.3.2.2 Analyses of Core Debris Coolability

In SECY-93-087, it was concluded that evolutionary light water reactors should ensure that the best estimate environmental conditions (pressure and temperature) resulting from core-concrete interactions do not exceed ASME Service Level C limits for steel containments or factored load category for concrete containments, for approximately 24 hours. In addition, designers should ensure that the containment capability has a margin to accommodate uncertainties in the environmental conditions from core-concrete interactions. Twenty-four hours is considered by the USNRC to be an appropriate time period based on sufficient time to allow for decay of fission products, operator intervention, utilization of accident management strategies, fission product deposition in the containment through natural mechanisms, and offsite protective measures. It was developed as a guideline and not a strict criterion in recognition of the uncertainties in severe accident progression and phenomenology.

TPC discussed debris coolability in Section 19.4.3.2.2.1 of the Lungmen PSAR in terms of a decomposition event tree but no results of the quantification of this tree were presented in this section. A more detailed discussion was provided in Appendix AJ.12 which was outside the scope of this review. Appendix AJ discusses the applicability of experiments to the ABWR and notes that MAAP3.0B-ABWR was selected as the deterministic model to evaluate CCI. The base case selected for the investigation of CCI was the low pressure loss of injection scenario. For this scenario, MAAP-ABWR determined that approximately 60 % of the total core inventory was released from the vessel to the drywell floor. Additional cases were run assuming that 100 % of the core debris was discharged from the vessel.

The following conclusions were reported in Appendix AJ:

- For all sequences with successful operation of the flooder, radial concrete erosion was less than the specified structural limit. Radial attack does not pose a significant challenge to the containment.

- For sequences with operation of the COPS, the fission product release is dominated by noble gas (due to suppression pool scrubbing).

- Release times for cases with the passive flooder are on the order of 20 hours after the onset of melting.

- The extended time period between vessel breach and rupture disk actuation (or containment failure) provides for a substantial reduction in the amount of fission products released from the containment.

- Using experimentally based values for upward heat transfer would result in debris cooling in the ABWR and early termination of CCI. Therefore, the lower bound for upward heat transfer is conservatively assumed to be 100 kW/m^2.

- For the dominant scenarios with successful operation of firewater to provide water to the debris, the time from the onset of core melting to fission product release is 24 hours from the beginning of the accident for all upward heat transfer rates.

In NUREG-1503, it was noted that GE considered accident classes representing the largest frequencies in selecting the accident sequences to be studied for the certification of the Standard ABWR. Eight accident sequences were selected for analysis using MAAP-ABWR. These accident sequences include loss of core cooling with the reactor vessel failing at low and high pressure, SBO, loss of containment heat removal, large break loss-of-coolant accident, and ATWS at low ant high pressure and ATWS concurrent with an SBO. For each accident sequence, several mitigating systems could be used to prevent or reduce the release of fission products into the environment. These mitigating systems include in-vessel recovery, passive flooder system, ACIWA, containment heat removal, and containment sprays.

The results of the GE analyses were reported to indicate core debris coolability and little, if any, CCI. The time-to-release of fission products ranges from 8.6 to 50 hours from the

start of the transient with the most likely fission product release location through the COPS. The COPS prevents the containment pressure from reaching the ASME Service Level C limit. However, for some sequences, the time to COPS actuation is less than 24 hours. For example, the accident sequence resulting in a release time of 8.6 hours is of extremely low probability involving an SBO with failure of the combustible gas turbine concurrent with an ATWS in which all reactivity control fails. However, if credit is given to operation of the ACIWA in the containment spray mode, the time-to-release of fission products increases to 26.4 hours.

NUREG-1503 also notes that a benchmark of the a containment's passive pressure capability is its ability to accommodate the loss of containment heat removal sequence analyzed by GE in which reactor vessel injection is maintained with all the decay heat being transferred to the suppression pool. Core damage does not occur. The GE analysis indicates that the COPS would actuate in approximately 21.7 hours. COPS actuation results from saturation of the suppression pool pressurizing the containment. This sequence indicates that the time to COPS actuation, even without the added pressurization and energy sources from severe accidents, cannot be extended much beyond 20 hours in the absence of active decay heat removal. With the addition of noncondensible gases from CCI and heat from the exothermic metal-water reactions during a severe accident, the time to COPS actuation will be less.

As noted in NUREG-1503, this is an important point in that COPS actuation before 24 hours cannot be prevented unless additional heat capacity is added to the containment or a containment heat removal system is recovered. The ACIWA system can provide additional heat capacity to prolong the time to COPS actuation. Based on GE's analysis for the ABWR SSAR, the time to COPS actuation is delayed by at least 10 hours for cases in which additional water is added to the containment by the ACIWA, when compared with the same sequence in which only the LDF system actuates to cool the core debris. The ACIWA is crucial to delaying the time to COPS actuation.

NUREG-1503 also noted that uncertainty analyses were performed by GE for the ABWR SSAR using MAAP-ABWR to investigate the uncertainties associated with debris coolability. These analyses evaluated the impact of parameters such as the amount of core debris, debris-to-water heat transfer, amount of steel in the debris, delayed flooding of the lower drywell, and use of the ACIWA system on CCI, containment pressurization,

COPS actuation, and fission product release. As discussed in Section 19.3.3.2.1.4 above, the Lungmen NPS will have a 1.5 m layer of basaltic concrete above the containment liner. This concrete layer is designed to protect the containment liner from being breached in the event that significant CCI occurs. The GE results provided in the ABWR SSAR, indicate that axial ablation will not exceed 1 m in a 24-hour period.

As discussed in Section 19.3.3.2.1.5 above, the distance the molten corium must ablate to result in pedestal structural failure in the radial direction is greater than 1.5 m. As reported in Table AJ.12-2 of the Lungmen PSAR, MAAP-ABWR estimates of the radial attack indicate that radial ablation does not represent a significant threat to the Lungmen containment.

For the discussion of Case 3 on page 19.4-18 of the Lungmen PSAR, the argument does not appear to support the assignment of probabilities that is identified for the event tree. TPC is planning to perform sensitivity analyses in this regard for the FSAR.

In summary, based on the analyses reported in Appendix AJ.12 and the similarities between the Lungmen NPS and the GE Standard ABWR, it is concluded that the Lungmen meets the criterion in SECY-93-087 for ensuring that containment conditions do not exceed ASME Service Level C limits for approximately 24 hours from CCI when the use of mitigation systems incorporated into the design (such as the LDF and ACIWA) are given credit.

The ACIWA is fundamental to prolonging the period to COPS actuation. Once COPS is actuated, containment pressurization is relieved through a controlled pathway that takes advantage of scrubbing by the suppression pool.

19.3.3.2.3 Basis for Acceptability

The Lungmen NPS appears to satisfy the criteria of SECY-93-087 for core debris capability by (1) providing a lower drywell unobstructed floor area greater than 79 m^2 to enhance debris spreading, (2) providing LDF and ACIWA systems to flood the lower drywell, (3) providing a sacrificial layer of basaltic concrete to protect the liner in the drywell floor, (4) providing a thick reactor vessel pedestal, and (5) providing the COPS to prevent containment structural failures due to long term containment pressurization. Actuation of the COPS results in a scrubbed, controlled release to the environment.

Given the above design features, containment conditions resulting from CCI can be maintained below ASME Service Level C conditions for approximately 24 hours.

In the discussion of the decomposition event tree for debris coolability in Section 19.4.3.2.2.1 of the Lungmen PSAR, there was no reference to the possibility that the debris might not spread uniformly over the entire drywell floor area. The impact of non-uniform distribution of the debris was excluded in the discussion because it was judged to be a second order effect. TPC has committed to adding a discussion in future revisions to explain the bases for this conclusion. The discussion will identify potential causes and configurations of non-uniform debris distributions, and the probability and effect of such distributions.

19.3.3.3 High-pressure Core Melt Ejection

High-pressure core melt ejection is the term used for the ejection of core debris from the reactor vessel when the reactor vessel pressure is high. Direct containment heating is the term used for the sudden heatup of the containment atmosphere and the associated pressurization resulting from the fragmentation and dispersal of core debris within the containment atmosphere. HPME and DCH are severe accident phenomena that could lead to early containment failure, resulting in a large release of radioactivity to the environment.

19.3.3.3.1 Preventive and/or Mitigative Features

In SECY-90-016 (Reference 19-2), it was concluded that evolutionary LWR designs should include a depressurization system to preclude HPME and cavity design features to contain ejected core debris if it cannot be precluded.

The Lungmen NPS has a reliable Automatic Depressurization System (ADS) to depressurize the reactor if high pressure injection systems fail, thus minimizing the frequency of accident sequences involving high pressure at the time of vessel breach. The ADS is used to depressurize the reactor vessel when it is shut down and isolated. The ADS consists of 8 SRVs, a subset of a total of 18 SRVs. The SRVs provide three main functions: overpressure relief operation using pneumatic actuators, safety operation using steam overpressure, and depressurization using the ADS valves. The

eight ADS SRVs can function in either the ADS or SRV mode. All of the SRVs are located on the main steam lines and discharge to the suppression pool.

The ADS valves require dc electrical power for the solenoid valves and a nitrogen gas supply for the servo valves and pneumatic actuators. The ADS SRVs have two accumulators, one of which is dedicated to the ADS function. Nitrogen gas is supplied from either the high-pressure nitrogen gas supply system or two backup safety-grade nitrogen gas supplies through an accumulator. The SRVs of the ADS each have two accumulators: one for the ADS function and one for the relief function. The accumulator capacity is sufficient for one actuation at drywell design pressure or five actuations at normal drywell pressure. The ADS is initiated either automatically or manually.

The ADS valves must remain open until the time of vessel breach if they are to prevent HPME. Once the reactor vessel fails, the ADS is no longer needed. Section 19.4.3.7.2.1.1 of the Lungmen PSAR notes that for the 10CFR50.34(f) and in-vessel scenarios, the drywell temperature and pressure never exceed the SRV design limits. For scenarios that proceed to the ex-vessel phase, the drywell temperature and pressure remain below the design limits until the vessel depressurizes. Thus, the severe accident conditions of interest do not pose a problem relative to SRV survivability.

Given that vessel breach occurs at high pressure, the design of the Lungmen lower drywell is such that the transfer of ejected core debris to the upper drywell is inhibited by the following mechanisms: (1) debris capture and trapping in the lower drywell, (2) impaction and removal of debris as it transported between the lower and upper drywell, and (3) partitioning of the debris and gas flows exiting the lower drywell region between the upper drywell and the wetwell. The upper portion of the lower drywell region contains the CRD mechanisms. Debris entrained in the lower drywell atmosphere may become trapped on these mechanisms and other stagnant areas within the lower drywell. For debris to be transported from the lower drywell floor region to the upper drywell, it must first travel vertically to the height of the downcomer vent, make a 90-degree turn into the vent, travel horizontally through the vent line, and then make a second 90-degree turn before it travels vertically through the downcomer vent to the upper drywell. This is the only existing pathway between the lower and upper drywell regions. This relatively tortuous and indirect pathway is expected to enhance the removal of core debris from the gas stream by impaction. As the pressure in the lower

drywell increases, the level of the suppression pool water within the lower regions of the downcomers will be forced down to expose the horizontal vents to the suppression pool. Once the suppression pool vents have cleared, the gas and debris exiting the lower drywell region will be partitioned between two paths: one to the upper drywell and one to the wetwell. The potential pathway around the reactor vessel is closed off by the reactor vessel skirt. This prevents core debris from the lower drywell from being ejected alongside of the vessel into the upper drywell.

Because the Lungmen containment is inerted, any pressure and temperature loading due to hydrogen combustion will be limited by the relatively small amount of oxygen that is available.

19.3.3.3.2 Basis for Acceptability

Evolutionary LWR designs must provide a reliable depressurization system and cavity design features to decrease the amount of ejected core debris that can relocate to the upper containment. The Lungmen ADS is designed with a reliable nitrogen supply and dc power to ensure its operability. The Lungmen containment design contains features that inhibit the transfer of ejected core debris from the lower drywell to the upper drywell. Therefore, it is concluded that the criteria of SECY-93-087 for high pressure melt ejection have been met.

19.3.3.4 Containment Vent Design

The Lungmen NPS has a COPS that is intended to provide protection against low frequency accident scenarios in which containment structural integrity is challenged by overpressurization. TPC has estimated that the scenarios involving potential overpressurization represent only 13 % of the hypothesized severe accident sequences. In the absence of the COPS, unmitigated overpressurization of the containment will result in failure of the drywell head for most severe accident scenarios. Failure of the drywell head could result in fission product releases that have not been scrubbed. The COPS provides a controlled scrubbed vent path from the wetwell with provisions that allow for reisolation of the containment.

The desirability of venting a BWR containment to mitigate beyond-the-design-basis accidents has been accepted for some time. Since 1981, the BWR EPGs, developed by the BWR Owners' Group and approved by the USNRC for existing BWRs, have called for venting the containment wetwell airspace. Subsequently, all US operating plants with Mark I containments have installed hardened vents for containment pressure relief capability in response to Generic Letter 89-16 issued by the USNRC.

The Lungmen NPS has a COPS that is designed to avoid gross containment failure resulting from postulated slow-rising overpressure scenarios. The COPS vents the containment from the wetwell airspace, thereby taking advantage of the scrubbing capability of the suppression pool. Any subsequent fission product release to the environment will be greatly diminished by the scrubbing provided by the suppression pool.

19.3.3.4.1 System Description

The COPS is part of the ACS and consists of two 200 mm diameter overpressure relief rupture disks and two normally open air-operated isolation valves installed in series in a 250 mm line that connects the wetwell air space to the plant stack. The isolation valves are located upstream of the rupture disks, as close as practical to the containment and provide the capability to reestablish containment integrity subsequent to venting. Once containment pressure has been reduced to a permissible level and containment heat removal capability has been reestablished, the two normally open COPS isolation valves will be manually closed.

The setpoint of the inboard rupture disk is selected to vent the wetwell airspace when its pressure reaches 0.72 MPa absolute. The outboard rupture disk, located at the inlet to the plant stack, fails at 0.03 MPa differential pressure. The second rupture disk setpoint is set much lower than that of the first rupture disk to eliminate the possibility of pressurization between the rupture disks adversely affecting the COPS actuation pressure. If leakage were to occur between two high-pressure rupture disks, the containment pressure required to burst the first rupture disk could exceed the design bursting pressure of the disk. By utilizing a high-pressure rupture disk in succession with a low-pressure rupture disk, it is expected that significant in-leakage would pressurize the airspace between the disks and burst the low-pressure rupture disk, thereby not

affecting the required containment pressure for bursting the high-pressure rupture disk. The piping volume between the rupture disks is inerted with nitrogen to eliminate the potential for combustion within the portion of the vent path located within the reactor building. Nitrogen at a pressure below the setpoint of the outboard rupture disc will be used to inert the COPS discharge piping between the rupture discs. The nitrogen pressure will be controlled and monitored during inerting to ensure that the pressure in the discharge piping is within specified limits. The specific test procedure to ensure that the disk is intact after inerting will be defined in the FSAR.

TPC notes that several factors were considered in determining the optimum setpoint for the rupture disk. The setpoint of the rupture disk must be optimized for the competing factors of minimizing the probability of drywell head failure and maximizing the time before fission products are released to the environment. It was concluded that the rupture disk setpoint should be set at 0.72 MPa absolute (at 93 °C), which is lower than the drywell head limited ASME Service Level C goal of 0.77 MPa absolute. The COPS setpoint of 0.77 MPa absolute is estimated by TPC to result in a 5 % probability of drywell head failure prior to rupture disk opening for a maximum anticipated drywell-to-wetwell pressure difference of 0.1 MPa.

The purpose of the COPS isolation valves is to contribute to the control of the venting process. Following rupture disk actuation, plant operators may decide to reclose the vent path based upon accident management guidance or procedures to be developed by TPC. The COPS isolation valves are designed to be capable of fully opening and closing against the pressures associated with venting. In Section 19.4.3.4.3 of the Lungmen PSAR, it is noted that the sizing of the COPS vent path is sufficient to allow 35 kg/sec of steam flow at the rupture disk opening pressure of 0.72 MPa absolute. This flow rate corresponds to an energy flow of approximately 2.4 percent rated power. TPC notes that for virtually all severe accident sequences, the rupture disk would not be actuated until approximately 20 hours after scram. The decay heat level at 20 hours after scram is approximately 0.5 % of full power. Thus, it is concluded that sufficient margin exists to ensure pressure relief once the rupture disk has been actuated.

TPC also performed calculations to investigate the potential effects of suppression pool swell and associated fission product carryover at the time of COPS actuation. The TPC analyses are reported to indicate that suppression pool swell does not threaten the

integrity of the COPS and that no significant entrainment of fission products will occur due to carryover.

The isolation valves in the COPS pathway are subject to the leak-testing requirements associated with 10 CFR Part 50, Appendix I. As indicated above, these valves are normally open so that the containment atmospheric pressure is exposed to the first rupture disk during all modes of operation. The COPS isolation valves are designed to fail-open upon loss of actuating power. This fail-safe position ensures venting via the COPS during severe accident scenarios involving multiple failures. If leakage past the rupture disks occurs under design basis conditions, the valves can be remotely isolated from the control room. In-line radiation monitoring of the vent pathway could be used to detect leakage. However, no leakage is expected at containment pressures less than the containment design-basis (0.31 MPa gauge).

The rupture disk setpoint was selected to assure an adequate margin prior to drywell head failure while maximizing the time before fission product releases through COPS. The rupture pressure of the COPS is slightly below the containment's ASME Service Level C capability of 0.77 MPa absolute which provides confidence that the integrity of the containment will be maintained before rupture disk actuation.

19.3.3.4.2 Basis for Acceptability

In SECY-90-016, it was recommended that the use of an overpressure protection system that used a dedicated vent system for the GE ABWR be approved. The USNRC subsequently approved the recommended use of the COPS on the ABWR, subject to the results of a comprehensive regulatory review, which should fully weigh the potential "downside" risks with the mitigation benefits of the system. In addition, the USNRC directed its Staff to ensure that the design should provide full capability to maintain control over the venting process.

19.3.3.4.2.1 Review of the Downside Risks and Mitigation Benefits

The COPS is intended to protect the containment during accident sequences in which containment integrity is challenged by overpressurization. Without the COPS, the containment failure location is expected to be the upper drywell head with a mean

ultimate pressure capability of 1.025 MPa absolute (at 260 °C). COPS actuation is clearly preferable to failure of the upper drywell head for the following reasons: (1) fission product releases through the COPS will be scrubbed by the suppression pool, and (2) the COPS vent pathway can be isolated when desired. The actuation setpoint and relief capacity of COPS have been optimized to minimize unnecessary releases while, at the same time, providing adequate margin to prevent containment failure due to overpressurization.

The potential downside risks of COPS relate to the containment failure frequency and source term. The pressure setpoint for COPS actuation has an impact on the containment failure frequency. If the setpoint is too high, the potential exists for failure of the upper drywell head before COPS actuation. For NUREG-1503, GE evaluated the variability in the pressure setpoint at which COPS is actuated for the standard AWBR, as well as the uncertainty of the drywell head failure pressure. Based on this evaluation, GE concluded that there was between a 2 percent and 5 percent probability of drywell head failure before COPS actuation, dependent upon the specific accident scenario.

The temperature of the rupture disks has an impact on its bursting pressure. Higher temperatures cause a decrease in the bursting pressure, whereas lower temperatures cause an increase. NUREG-1503 reported that, for the Standard ABWR, GE had evaluated the range of time at which the rupture disks would fail for a range of temperatures from 38 to 149 °C. GE concluded that for the range of the temperatures evaluated, the time to rupture disk failure was within 0.8 hours of the base case and the probability of drywell head failure before rupture disk failure varied slightly. These results indicate that rupture disk temperature variations in the range of interest have only a minor effect on the ability of COPS to mitigate containment overpressurization.

A potential adverse impact of the COPS (i.e., increased risk) results from its actuation when it might have been possible to recover a containment heat removal system in the time period after COPS initiation and before failure of the upper drywell head. The recovery of the containment heat removal system would prevent containment pressure from reaching the rupture disk setpoint, thereby preventing a fission product release through the COPS. In Section 19.4.3.4.4 of the Lungmen PSAR, TPC estimated the probability of recovering containment heat removal capability between the predicted

times of rupture disk and drywell head failure (in the absence of rupture disk failure) to be approximately 4 to 11 percent, depending upon the accident scenarios.

A comparison of the Lungmen containment performance with and without the use of the COPS was presented in Section 19.4.3.3.4 of the Lungmen PSAR. This included comparisons of the time and magnitude of fission product release for the frequency-dominant sequence for the Lungmen NPS, as well as an assessment of the impact of COPS on the frequency of core damage and drywell head failure for various accident classes. These results indicate that for the dominant sequence, COPS reduces the time of fission product release from approximately 35 to 31 hours if drywell sprays are available. For cases with passive flooder operation, the release of fission products through the COPS occurs approximately 6 to 8 earlier than the release associated with drywell head failure.

MAAP-ABWR calculated Cesium Iodine (CsI) release fractions at 72 hours for various accident scenarios ranged from 0.3 to 37 % when the release was related to a failure of the drywell head. For all cases where the release was through the COPS, the calculated CsI release fractions were of the order of 1E-7.

In Section 19.4.3.4.7 of the Lungmen PSAR, it is noted that the probability of drywell head failure would increase by a factor of 50 to 100 in the absence of COPS.

In summary, the major benefit of COPS is that it provides assurance that releases from the Lungmen containment are scrubbed by the suppression pool. Operators can manually control releases via COPS via air-operated isolation valves. While actuation of COPS results in an earlier release of fission products relative to the time of failure of the upper drywell head, the magnitude of the release is significantly less. COPS is also expected to reduce the frequency of core damage resulting from accident sequences involving loss of core cooling induced by containment failure.

As noted earlier, a major concern related to COPS is that it could lead to earlier, and perhaps unnecessary, releases. However, this does not seem to be a significant factor for the Lungmen NPS. In particular, the selection of the system setpoint appears to provide a reasonable balance between the competing goals of minimizing the probability of drywell head failure and maximizing the time before fission products are released to the environment. The setpoint is sufficiently high that the time of release will generally

still be on the order of 15 to 20 hours, yet low enough that the probability of containment failure before COPS actuation is relatively small (about 5 %). Also, the probability of unnecessary system actuation (when it might have been possible to prevent a release by recovering containment heat removal systems before containment failure) appears to be reasonably low (between 4 and 11%) and COPS is capable of being manually isolated from the control room.

Based on the information presented in the PSAR, it is concluded that the inclusion of COPS in the Lungmen containment results in a significant net risk reduction.

19.3.3.4.2.2 Control Over the Venting Process

The provisions provided by TPC for ensuring full capability to maintain control over the venting process are as follows:

a) selection and justification of the containment pressure when the COPS will actuate.

b) provisions for reclosure of the vent pathway using containment isolation valves (CIVs) that are designed to be capable of fully opening and closing at pressures up to the COPS actuation pressure.

c) providing radiation monitoring within the vent pathway to monitor the potential for an early release or provide guidance for accident management strategies for reclosure of the vent pathway.

d) sizing of the vent pathway to prevent further containment pressurization, once the venting process is actuated, and prevent suppression-pool-level swell into the system piping.

The use of a rupture disk in the COPS pathway prevents the opportunity for venting at pressures below the actuation pressure of the rupture disks. It is believed that a good engineering rationale can be developed to support either position--to allow for early venting or to maintain containment integrity as long as possible. The decision to provide rupture disks and therefore prevent early venting is philosophical in nature, with the underlying theme of maintaining an intact containment and venting only as a last resort.

The containment is the final barrier preventing release of radioactivity into the environment. As such, it should not be unnecessarily or prematurely breached by plant staff.

19.3.3.4.3 Conclusions

The Lungmen NPS dedicated containment vent is the COPS, which consists of a vent path from the wetwell airspace to the plant stack. The COPS actuates at a pressure less than ASME Service Level C limit and has provisions for isolation following actuation. Control over the venting process is assured through selection of the actuation pressure and the capability for vent path reclosure. The results of the review indicate that the net benefits of the system outweigh the potential negative aspects. Based on the above, it is concluded that the Lungmen NPS design meets the criteria specified in SECY-90-016 for inclusion of a dedicated containment vent path.

19.3.3.5 Suppression Pool Bypass

Although suppression pool bypass is not a severe accident phenomenon, it can have a significant impact on accident source terms and can be a significant contributor to plant risk. Suppression pool bypass is associated with the failure of the containment system to channel steam and fission product releases through the suppression pool.

The fundamental characteristic of a BWR pressure-suppression containment is that steam released from the RCS will be condensed, limiting the pressure rise in the containment, and scrubbed of radionuclides in the suppression pool. These characteristics are accomplished by directing steam and gases from the RCS to the suppression pool through a drywell to wetwell connecting vent system. However, leakage paths could exist in the pathway between the drywell and wetwell that would allow steam to bypass the suppression pool, pressurize the containment, and lead to a release. Potential sources of steam bypass include leakage through the vacuum relief valves, cracking of the drywell structure, and penetrations through the drywell structure.

To address the importance of suppression pool bypass, the potential mechanisms, probabilities and source locations were reviewed by TPC to identify where fission products might be released outside of the containment. The TPC analysis

conservatively focused on the station blackout event since it leads to a higher likelihood of suppression pool bypass and because it is considered one of the more probable initiating events for core damage sequences.

19.3.3.5.1 Preventive and/or Mitigative Features

In SECY-90-016, it was concluded that a special effort should be made to eliminate or further reduce the likelihood of a sequence that could bypass the containment. In SECY-93-087, it was stated that vendors should make reasonable efforts to minimize the possibility of bypass leakage and should account, in their containment designs, for a certain amount of bypass leakage.

In Section 19.4.3.5.2.1.1 of the Lungmen PSAR, TPC concludes that the only mode of suppression pool bypass that presents any significant risk during a severe accident is vacuum breaker leakage. LOCAs outside containment and containment isolation failures were examined but were not deemed to be important contributors to risk. The Lungmen NPS contains 8 vacuum breakers. These vacuum breakers are essentially swing check valves that begin to open passively when the wetwell pressure exceeds the drywell pressure by 0.0014 MPa and are fully open when this pressure difference reaches 0.0045 MPa. If the vacuum breaker disk fails to reseat, a bypass of the suppression pool would be created and allow steam in the drywell to pass directly to the wetwell airspace. Such a scenario bypasses the condensation function of the suppression pool and challenges containment integrity. For sufficiently large amounts of leakage during a severe accident, the time to rupture disk opening or containment failure can be significantly reduced and the magnitude of the fission product release can be increased.

To address the impact of vacuum breaker leakage, TPC performed a study using decomposition event trees and deterministic modeling. The decomposition event tree shown in Figure 19.4-19 of the Lungmen PSAR indicated that given an event that resulted in vacuum breaker operation, the probability of creating a small bypass was 0.018 and the probability of creating a large bypass was 0.0039. The probability of a vacuum breaker sticking open or leaking will depend on the number of cycles they experience during any given accident. The TPC analysis focused on the station blackout event because it leads to a higher likelihood of suppression pool bypass and because it was considered one of the more probable initiating events for core damage sequences.

To mitigate the consequences of suppression pool bypass for both DBAs and severe accidents, the Lungmen NPS design includes a containment spray system in the wetwell and upper drywell. The RHR system supplies water to the safety-grade sprays. Although the RHR pumps are automatically aligned for their ECCS function, the flow can be manually diverted to the containment sprays to mitigate the pressurization of the wetwell airspace during suppression pool bypass.

In the event that the RHR system fails, the ACIWA system can be interconnected with the RHR system to provide flow to the upper drywell containment spray header. The COPS prevents gross containment failure for bypass scenarios in which the containment pressure would exceed the rupture pressure of the rupture disk. COPS allows for a controlled release through reclosure of isolation valves in the vent pathway.

19.3.3.5.2 Analyses of Suppression Pool Bypass

In Section I9.4.3.5.2.1.3 of the Lungmen PSAR, TPC discussed the results of analyses that had been performed to determine the impact on the containment from varying amounts of suppression pool bypass. The analysis evaluated the effects of varying a vacuum breaker's bypass leakage area from zero leakage to leakage from one full open vacuum breaker (2030 cm^2) on the time to fission product release and the Cesium Iodine release fraction at 72 hours for the following five different scenarios:

1. Bypass leakage begins after passive flooder activation; aerosol plugging is neglected.

2. Bypass leakage is present from the beginning of the accident; aerosol plugging is neglected.

3. Bypass leakage begins after passive flooder activation; aerosol plugging of the vacuum breaker opening is considered.

4. Bypass leakage is present from the beginning of the accident; aerosol plugging of the vacuum breaker opening is considered.

5. Bypass leakage is present from the beginning of the accident and the operator initiates the firewater spray system (ACIWA).

The time to fission product release and the CsI release fractions were determined using MAAP-ABWR runs. This analysis used the aerosol plugging model developed by Vaughan (Reference 19-14). The dominant severe accident sequence, LCLP (loss of all core cooling with vessel failure occurring at low pressure), was chosen to evaluate plant performance.

MAAP-ABWR predicts that the release fraction of CsI for the LCLP case without bypass leakage is less than 1.0E-7. The effect of leakage on the CsI fraction is indicated in the following table:

Amount of Leakage	CsI Release Fraction f
None	$f < 1.0E-7$
Small	$0.01 < f < 0.1$
Large	$f > 0.1$

As indicated above, CsI release fractions are orders of magnitude larger for cases with bypass leakage than for the case without bypass leakage. According to Lungmen PSAR Table 19.4-4, for cases with effective suppression pool bypass areas greater than 400 cm^2, the 72-hour CsI release fractions are greater than or equal to approximately 0.16.

The effect of leakage on time to rupture disk opening is shown in the following table:

Amount of Leakage	Time to Rupture Disc Opening (hrs)
None	~ 20
Small	6 to 16
Large	< 6

Scenario 5 examined the effects of drywell spray from the ACIWA on cases with bypass leakage present from the beginning of the accident. Assuming the operator initiates the

firewater spray within 2 hours of the start of the accident, Table 19.4-4 indicates that the elapsed time to rupture disk opening can be delayed to nearly 30 hours for bypass pathways experiencing up to one fully open vacuum breaker.

19.3.3.5.3 Basis for Acceptability

The Lungmen NPS has a containment spray system and COPS to mitigate the effects of containment bypass and prevent a bypass scenario from progressing to containment failure. The containment spray system can be supplied from either the RHR or ACIWA system. In SECY-90-016, it was stated that venting should be delayed for approximately 24 hours following the onset of core damage. For the cases in which one vacuum breaker is fully open, the Lungmen NPS meets the intent of this criterion if containment spray (via the ACIWA system) is initiated within 2 hours (as described in scenario 5 above).

It is concluded that TPC has performed a relatively complete analysis to allow an understanding of the capability of the Lungmen NPS containment to accommodate a range of bypass conditions through the vacuum breakers. This analysis highlights the importance of the containment spray system to mitigating the consequences of suppression pool bypass. In addition, the probability of vacuum breaker leakage has been estimated to be only approximately 2 %. Thus, suppression pool bypass does not appear to significantly add to the risk associated with the Lungmen NPS.

19.3.3.6 Fuel-Coolant Interaction

The containment function can be challenged by energetic or rapid energy releases such as the energy release associated with an FCI that results in a steam explosion. The term "steam explosion" refers to a phenomenon in which molten fuel rapidly fragments and transfers its energy to the coolant resulting in rapid steam generation, expansion, shock waves, and possible mechanical damage. To be a significant safety concern, the interaction must be very rapid and must involve a large fraction of the core mass. Steam explosions can occur either in-vessel or ex-vessel.

19.3.3.6.1 In-Vessel Steam Explosion

NUREG-1116 (Reference 19-15) summarized the deliberations, from the first meeting, of the Steam Explosion Review Group's (SERG's) understanding of the potential for containment failure arising from in-vessel steam explosions during core melt accidents. The consensus reached by the SERG was that the occurrence of an in-vessel steam explosion of sufficient energetics to lead to containment failure (the so-called alpha-mode of failure) was sufficiently low in probability to allow its elimination as a credible threat. Despite differences in opinion relative to the modeling of basic steam explosion sequence phenomenology, an opinion supported by most members of the group was that the probability of containment failure is reduced for one or more of the following reasons:

- limited melt mass involvement in the explosion.
- low thermal-to-mechanical energy conversion.
- incoherent relocation of the molten corium due to the existence of structures (such as the CRD guide tubes) in the lower reactor vessel head.

A second SERG workshop was held in 1995 to review the status of fuel-coolant interaction research. It was again concluded that the overall likelihood of early failure from this challenge is low and this issue was resolved from a risk perspective (Reference 19-16). The results of this review meeting were documented in NUREG-1524 (Reference 19-17).

Based on the preceding conclusions, TPC judged that the risk of containment failure by in-vessel steam explosions was insignificant and did not warrant further analysis.

19.3.3.6.2 Ex-Vessel Steam Explosion

In SECY-93-87, it was stated that any dynamic forces resulting from ex-vessel FCI on the integrity of the containment should be evaluated. One of the conditions necessary for an ex-vessel FCI in the Lungmen NPS is for the molten corium to be discharged from the reactor vessel into an existing body of water in the lower drywell. The design of the

Lungmen NPS containment substantially reduces the probability of a preexisting body of water in the lower drywell at the time of reactor vessel failure.

The reactor vessel skirt is solid, preventing the transfer of water from the upper drywell to the lower drywell. In addition, there are no active injection systems in the lower drywell, and the passive flooder system does not actuate until the atmosphere within the lower drywell is approximately 260 °C. The connecting vent between the upper and lower drywell regions is a vertical channel which has a horizontal branch leading to the lower drywell. In order for water flow from upper drywell to enter the lower drywell, the water level in the wetwell would have to rise sufficiently to overflow into the connecting vent. If the EPGs are followed, this overflow could only occur if injection was being provided from an external source in the event that flow from the suppression pool was not available.

In Section 19.4.3.6.2.1 of the Lungmen PSAR, TPC stated that the total frequency of the core damage sequences that result in water on the lower drywell at the time of vessel failure is equivalent to 0.3 % of the total core damage frequency. The low likelihood of a flooded lower drywell at the time of reactor vessel failure provides a sufficient basis to conclude that the probability of an ex-vessel steam explosion has been reduced to an acceptably low value and is therefore acceptable. However, TPC provided a discussion of the experimental database and its relevance to the Lungmen NPS configuration as well as a discussion of a simple, scoping calculation that was performed to estimate the ability of the Lungmen containment to withstand a large, energetic fuel coolant interaction.

19.3.3.6.2.1 Analysis

The analysis reported in Section 19.4.3.6.3.4.1 of the Lungmen PSAR, based on simple elastic-plastic calculations, indicated that the pedestal can withstand a maximum pressure of 1.7 MPa and that the conservative lower limit for pedestal failure is 0.85 MPa. The amount of corium that must participate in a FCI to achieve this lower limit is 22,400 Kg (Figure 19.4-22 of the Lungmen PSAR). The Lungmen NPS reactor vessel can contain 235,000 Kg of corium. Thus, the pedestal can withstand a FCI involving approximately 9.5 % of the corium inventory. TPC notes that this amount of corium is more than 3 times the maximum amount of debris that could participate in an FCI event

based on observations discussed in Section 19.4.3.6.2.2 of the Lungmen PSAR. Thus, it was concluded that the pedestal is very resistant to the impulse loading that could occur during a severe accident.

19.3.3.6.3 Basis for Acceptability

Based on the conclusions reached in NUREG-1524, it is concluded that in-vessel steam explosions are not a threat to the Lungmen NPS containment.

As discussed above in the previous section, the design of the Lungmen NPS containment substantially reduces the probability of a preexisting body of water in the lower drywell at the time of reactor vessel failure. The low likelihood of a flooded lower drywell at the time of reactor vessel failure provides a sufficient basis to conclude that the probability of an ex-vessel steam explosion has been reduced to an acceptably low value and is therefore acceptable.

19.3.3.7 Equipment Survivability

The purpose of this section is to discuss the survivability of equipment, both electrical and mechanical, that is needed to prevent and mitigate the consequences of severe accidents.

In the certification of the Standard ABWR, the USNRC's position is that features provided only (not required for design basis accidents) for severe-accident protection (prevention and mitigation) need not be subject to the 10 CFR 50.49 environmental qualification requirements; 10 CFR Part 50, Appendix B quality assurance requirements; and 10 CFR Part 50, Appendix A redundancy/diversity requirements. Accordingly, the Standard ABWR must include analyses, based on best-available methods, to demonstrate that:

- Equipment, both electrical and mechanical, needed to prevent and mitigate the consequences of severe accidents is capable of performing its function for the time period needed in the best-estimate environmental conditions of the severe accident (e.g., pressure, temperature, radiation) in which the equipment is relied upon to function.

- Instrumentation needed to monitor plant conditions during a severe accident is capable of performing its function for the time period needed in the best-estimate environmental conditions of the severe accident (e.g., pressure, temperature, radiation) in which the instrumentation is relied upon to function.

TPC addressed equipment survivability in Section I9.4.3.7 of the Lungmen PSAR and noted that the requirements are derived from two sources; 10 CFR50.34(f) and SECY-09-016.

The applicable criterion for equipment, both mechanical and electrical, required for recovery from in-vessel severe accidents is provided in 10 CFR 50.34(f).

Part 50.34(f)(2)(ix)(C) states that equipment necessary for achieving and maintaining safe shutdown of the plant and maintaining containment integrity will perform its safety function during and after being exposed to the environmental conditions attendant with the release of hydrogen generated by the equivalent of a 100 percent fuel-clad metal-water reaction including the environmental conditions created by activation of the hydrogen control system.

Part 50.34(f)(3)(v) states that systems necessary to ensure containment integrity shall be demonstrated to perform their function under conditions associated with an accident that releases hydrogen generated from 100 percent fuel-clad metal-water reaction.

Part 50.34(f)(2)(xvii) requires instrumentation to measure containment pressure, containment water level, containment hydrogen concentration, containment radiation intensity, and noble gas effluents at all potential accident release points.

Part 50.34(f)(2)(xix) requires instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

These regulations collectively indicate the need to perform a systematic evaluation of all equipment, both electrical and mechanical, and instrumentation to ensure its survivability for intervention into an in-vessel severe accident.

The applicable criteria for equipment, both electrical and mechanical, required to mitigate the consequences of ex-vessel severe accidents is discussed in the Equipment

Survivability section of SECY-90-016. This section indicates that features provided only (not required for design basis accidents) for severe-accident protection (prevention and mitigation) need not be subject to the 10 CFR 50.49 environmental qualification requirements; 10 CFR Part 50, Appendix B quality assurance requirements; and 10 CFR Part 50, Appendix A redundancy/diversity requirements. This results from the judgment that severe core damage accidents should not be treated in the same manner as design basis accidents in the traditional sense.

However, mitigation features must be designed to provide reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed. In cases where safety-related equipment (equipment provided for DBAs) is relied upon to cope with severe accident situations, there should be reasonable assurance that this equipment will survive accident conditions for the period that is needed to perform its intended function.

To meet the requirements for equipment survivability, TPC considered 3 categories of events. The first category considered one event that responds to the requirements of 10CFR50.34(f), paragraphs (2)(ix)(C) and (3)(v). A non-mechanistic scenario was considered that results in the oxidation of 100 % of the active fuel cladding. The other 2 categories respond to the requirements of SECY-90-016. The second category consists of events representing the frequency dominant events ending in in-vessel recovery and the third category is composed of events representing the frequency dominant events ending in ex-vessel recovery. TPC notes that the events in the second and third categories defined above represent 98 % of the core damage frequency.

For each of the 3 categories of events, a set of curves representing the bounding environmental conditions for that category were developed for use in evaluating the survivability of equipment and instrumentation. TPC then compared these conditions to the equipment capabilities to provide a measure of confidence that important equipment would survive the expected environmental conditions. Because the Lungmen containment is inerted, there will be no containment challenges due to hydrogen burns.

19.3.3.7.1 Events Ending in In-vessel Recovery

The Category 1 event evaluated for the equipment survivability issue involves a non-mechanistic event that results in 100 % clad oxidation but core cooling is recovered before vessel failure. Since the Lungmen core damage frequency is dominated by transient induced accidents, a transient initiated event was selected for this evaluation. Best estimate analyses do not result in 100 % clad oxidation. MAAP-ABWR models were conservatively adjusted and ECCS was cycled on and off to produce an amount of hydrogen equivalent to a 100 % clad reaction and the corresponding time-dependent environmental conditions were determined. The critical functions of reactivity control, vessel inventory control, containment isolation, and containment integrity were considered. However, not all of the equipment and instrumentation associated with these functions was evaluated. For example, the functions of reactivity control and containment isolation are required in the very early stages of the accident, during which all parameters are well within their design basis. Also, this Category 1 event does not impact equipment located in the secondary containment, and accordingly, it was not specifically evaluated.

By such screening, an equipment and instrumentation list was developed for each of the 3 categories of events. This list is provided in Table 19.4-7 of the Lungmen PSAR. The complete list of equipment important to severe accidents was derived from reviews of the safe shutdown equipment list, the EPGs, the Lungmen PRA, and the analysis of severe accidents.

The Category 2 events include the dominant in-vessel recovery sequences. These sequences are initiated by transients and involve a failure of core cooling. Representative environmental conditions were also developed by TPC for this category of events as well.

19.3.3.7.2 Ex-Vessel Severe Accidents

The category of events representing ex-vessel severe accidents is characterized by loss of core cooling, core melt and vessel failure. Containment response is characterized by pressurization due to steam and non-condensable gas generation. When the vessel fails, high temperatures are generated in the drywell for a short period of time due to the introduction of core debris into the lower drywell. Since events involving high pressure

at the time of vessel breach have different characteristics than those involving low pressure at the time of vessel breach, TPC generated two sets of bounding profiles.

As was the case for the other categories, TPC performed a detailed evaluation of what equipment and instrumentation was necessary and during what period of time. Differences in requirements for ex-vessel versus in-vessel accidents were highlighted. For the time period of interest, the bounding environmental conditions were compared to available qualification data.

19.3.3.7.3 Basis for Acceptability

As noted above, TPC developed sets of curves representing the bounding environmental conditions for both in-vessel and ex-vessel severe accidents. These environmental conditions were then compared to the equipment capabilities to provide a measure of confidence that the necessary equipment would survive the expected conditions. It is concluded that the systematic process used by TPC for assessing equipment survivability is acceptable and consistent with the assumptions used in TPC's deterministic assessment of severe accidents. It is also concluded that TPC's approach meets the requirements of 10 CFR 50.34 discussed in Section 19.3.3.7, and current and proposed applicable regulations for equipment survivability.

19.3.3.8 Protection of Containment Sumps

The Lungmen containment has two sumps located in the peripheral region of the lower drywell floor: an equipment drain sump (EDS) and a floor drain sump (FDS). These sumps could collect core debris during a severe accident if debris ingress to the sump is not prevented. An accumulation of core debris in these sumps could result in accelerated CCI and given the decreased thickness of concrete protecting the containment liner in the floor of the sumps, the time to liner melt-through due to CCI could be adversely affected. To prevent debris ingress and accumulation, and melt-through in the sump region due to CCI, corium shields will be provided around the periphery of the two sumps.

19.3.3.8.1 Sump Design Criteria

The following general design criteria for the drywell sumps were identified in Section 19.4.3.8.1 of the Lungmen PSAR:

- The shield wall material will have a melting temperature that is greater than the interface temperature between the debris and the shield wall.
- The shield height above the lower drywell floor shall be chosen to ensure long-term debris solidification and to prevent debris from collecting on top of the shield.
- The wall material chosen for the corium shields must have good chemical resistance to siliceous slags and reducing environments.
- The seismic adequacy of the corium shields will be determined in the detailed design phase.
- Sump shields have provisions to allow water to flow into the sumps when the lower drywell is flooded.
- The walls of the floor drain and equipment drain sumps extend to the floor of the sumps to prevent tunneling of debris into the sumps.

The EDS and FDS have somewhat different functions. Accordingly, some specific design criteria were also developed for these sumps. These criteria are identified in the sections that follow.

19.3.3.8.1.1 Equipment Drain Sump

The EDS collects water from known sources such as valve stem leakage. The water enters and exits through piping above the sump. The corium shield for the EDS will be solid except for the inlet and outlet piping which will go through its roof and its walls will be sufficiently thick to withstand ablation.

19.3.3.8.1.2 Floor Drain Sump

The FDS collects water leakage from unknown sources in the drywell. To facilitate the flow of this water into the FDS under normal conditions, flow channels through the corium shield wall must be provided. Thus, the following additional design criteria are specified for the FDS:

- The length of the channels in the FDS must be long enough to ensure that a plug forms in the channel before debris spills into the sump.
- The total flow area of the channels in the FDS corium shield wall shall be great enough to allow water flow rates stated in the Technical Specifications without causing excessive water pool formation in the lower drywell.
- The channel height shall be small enough to ensure freezing of the debris.

19.3.3.8.2 Corium Shield Design

Section 19.4.3.8.2 of the Lungmen PSAR discusses the corium shield design. In Section 19.4.3.8.1 of the Lungmen PSAR, it is suggested that alumina will be used for this protective layer of refractory bricks and that the height of the shield above the lower drywell floor will be 0.4 m. It is also noted that shield walls of the FDS and EDS extend to the floor of the sumps to prevent tunneling of debris into the sumps.

In Section 19.4.3.8.2.1 of the Lungmen PSAR, TPC presented the results of an analysis to determine the length of the channels in the FDS corium shield wall that will prevent molten debris ingress to the sump. A channel length of 0.5 m has been determined to satisfy this requirement. The channel height must be sufficiently small to ensure freezing of the molten corium. The TPC analysis was based on a channel height of 1 cm which satisfies this requirement.

The equations developed by TPC to assess the channel length required to ensure freezing were applied to 3 different melt scenarios. These scenarios differ in melt composition, material properties, initial temperature of the debris, and the ejection rate of debris from the failed vessel. The impact of debris superheat was addressed in a

sensitivity study (Lungmen PSAR Section 19.4.3.8.2.1.5.2). Scenarios I and II were taken from NUREG/CR-5423 (Reference 19-13). These scenarios represent predominantly oxidic and metallic melts, respectively. Scenario I is based on MAAP calculations that predict that there is a significant amount of molten debris available for relocation at the time of vessel failure. The debris release in this scenario is consistent with the core composition, corresponding to approximately 30 % by weight of zirconium. Scenario II as based on BWRSAR calculations that predict that debris which initially relocates to the lower plenum is quenched. Subsequent water depletion results in the remelting of the debris and local failure of the vessel. Since the metallic constituents of the debris remelt before the oxidic constituents, the initial debris pour from the vessel is primarily metallic in nature. Scenario III represents a MAAP-ABWR analysis that was used to predict a bounding debris ejection rate of primarily oxidic material from a failed vessel. These 3 scenarios are believed to cover the range of expected core melt conditions. Inserting the parameters associated with these 3 scenarios into the equations of interest all resulted in required channel lengths less than 0.5 m.

TPC performed sensitivity studies for important material properties and for the degree of superheat. These sensitivity studies supported the selection of 0.5 m as the required channel length.

The TPC analysis also addressed the requirements for shield walls without channels (Lungmen PSAR Section 19.4.3.8.2.2.3). TPC notes that the corium walls only need to be thick enough to provide a long-term stable interface between the debris on the lower drywell floor and the interior of the sumps. TPC determined that a thickness of 10 cm would provide suitable margin for any erosion due to initial debris contact with the wall.

19.3.3.8.3 Discussion

The sump shield designs proposed for Lungmen have considerable merit and some conservatism appears to be included in the design criteria. For example, the design criteria are intended to ensure that no core debris enters the sumps. The sumps, however, could withstand limited amounts of core debris.

Based on engineering judgment, it appears that the sump shields would prevent a substantial accumulation of core debris and that the channels within the FDS would lead

to freezing of debris within them. It is concluded that the sump shield design proposed by TPC is acceptable. This conclusion is reached based on the design criteria developed by TPC, the analysis that they performed, their evaluation of the shield design to variations in key parameters, and a review of existing related experimental and analytical work.

19.4 Containment Performance

The USNRC approach for ensuring containment survivability from severe accident challenges consists of requiring inclusion of accident prevention and consequence mitigation features and satisfaction of a containment performance goal (CPG). The CPG ensures that the containment would perform its function in the face of most severe accident challenges and that the design (including its mitigation features) would be adequate if called upon to mitigate a severe accident. Two alternative CPGs were identified in SECY-90-016: a conditional containment failure probability (CCFP) of 0.1 or a deterministic containment performance goal that offers comparable protection.

For internal events, it was concluded in NUREG-1503 that the Standard ABWR containment design limits the CCFP to approximately 0.1. The design detail for the Lungmen NPS containment was not available at the time the containment failure analysis was performed for the preparation of the Lungmen PSAR. As such, the Standard ABWR information was used in the containment analysis included in Appendix A to the Lungmen PSAR. However, it is expected that the Lungmen NPS containment is as capable structurally as the Standard ABWR containment. Based on the results of the analyses performed for the Standard ABWR, it is judged that the Lungmen NPS design should also limit the CCFP to approximately 0.1.

However, the use of a 0.1 CCFP was not imposed by the USNRC as a requirement. As noted in NUREG-1503, the USNRC Staff's requirement for the performance of GE Standard ABWR containment is as follows:

> *The standard design must include design features to limit the conditional containment failure probability for the more likely severe accident challenges.*

Table 19.5-1 of the Lungmen PSAR identifies the design features that mitigate severe accident phenomena and contribute to limiting the CCFP. The severe accident

phenomena that are mitigated by these design features are evaluated in Section 19.3.3 of this report. Based on the evaluations in Sections 19.3.3 of this report and the containment capacity analysis performed for the Lungmen NPS, it is concluded that the acceptance criteria in SECY-90-016, SECY-93-087, and the applicable regulation for containment performance have been met.

19.5 Severe Accident Management

The risk associated with severe accidents can be reduced significantly through severe accident management (SAM). SAM encompasses those actions taken during the course of a severe accident by the plant operating and technical staff to (1) prevent core damage, (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel, (3) maintain containment integrity as long as possible, and (4) minimize offsite releases. In SECY-88-147 and Generic Letter 88-20, the USNRC Staff identified a need for the development of an accident management plan by each operating reactor licensee for severe accident closure.

The Lungmen PSAR does not contain a severe accident management plan. However, TPC has committed to submitting a severe accident management plan as part of the FSAR. The staff will review the Lungmen SAM plan at the FSAR stage. TPC has stated that the plan will demonstrate its commitment to perform a systematic evaluation of the Lungmen plant's ability to deal with severe accidents in the unlikely event that they occur and to implement the necessary enhancements within the detailed plant design and organization.

19.6 Capacity Of Primary Containment

The pressure boundary of the Lungmen containment structure consists of the reinforced concrete containment vessel and the steel torispherical upper drywell head. Section 19.7 of Lungmen PSAR described TPC's commitments to the evaluations of containment internal pressure capability and associated failure modes and locations under severe accident conditions. During various severe accident conditions, the containment may be challenged by high temperatures from 533 K (260°C) to 644 K (371°C).

For the severe accident evaluation performed in the PSAR, TPC used the containment capacities calculated by General Electric for the Standard ABWR. TPC committed to use actual Lungmen specific containment capacities for the severe accident evaluation in the FSAR. This is acceptable to the staff at this stage of the application.

Evaluation methods and criteria for both service level C limits as well as median fragility levels for the Lungmen containment structure, including the reinforced concrete containment vessel and the upper drywell head, is to be provided in the FSAR. In addition, local failure modes and leak areas such as the interface of the upper drywell head and the top slab of the containment and leakage through fixed and operable penetrations will also be provided in the FSAR.

Since details for containment capacity evaluations and internal pressure capacities for various global and local failure modes will only be provided in the FSAR, the staff cannot perform a thorough review of the PSAR Section 19.7. Such information will be reviewed by the staff when it becomes available.

References:

19-1. USNRC, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," August 1985.

19-2. USNRC, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," SECY-90-016, January 1990.

19-3. USNRC, "Resolution of Selected Technical and Severe Accident Issues for Evolutionary Light Water Reactor (LWR) Designs," SECY-91-262, August 1991.

19-4. USNRC, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," SECY-93-087, April 1993.

19-5. USNRC, "Draft Standard Review Plan Chapter 19 – Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance," January 1998.

19-6. 10CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants."

19-7. 10CFR 50.63, "Loss of All Alternating Current Power."

19-8. 10CFR 50.44,"Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors."

19-9. USNRC, "Final Safety Evaluation Report Related to the Certification of the Advanced Boiling Water Reactor Design," NUREG-1503, July 1994.

19-10. Brookhaven National Laboratory, "Severe Accident Insights Report," prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-5132, April 1988.

19-11. Cronenberg, A.W., "In-Vessel Zircaloy Oxidation/Hydrogen Generation Behavior during Severe Acidents," prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-5597, September 1990.

19-12. Copus, E.R., et al, "Core-Concrete Interactions Using Molten UO_2 with Zirconium on a Basaltic Basemat," prepared for U.S. Nuclear Regulatory Commission, NUREG/CR-5564, August 1992.

19-13. Theofanous, T.G., et al, "The Probability of Liner Failure in a Mark-I Containment," NUREG/CR-5423, August 1991.

19-14. Vaughan, E.U., "Simple Model for Plugging of Ducts by Aerosol Deposits," Transactions of the American Nuclear Society, 28, 507, 1978.

19-15. USNRC, "A review of the Current Understanding of the Potential for Containment Failure From In-Vessel Steam explosions," NUREG-1116, June 1985.

19-16. USNRC, "Consideration of Severe Accident Risk in NRC Regulatory Decisions," SECY-97-171, July 1997.

19-17. USNRC, "A Reassessment of the Potential for an Alpha-Mode Containment Failure and a Review of the Current Understanding of Broader Fuel-Coolant Interaction Issues," NUREG-1524, August 1996.

19-18. USNRC, "Draft Safety Evaluation Report on the General Electric Boiling Water Reactor Design Covering Chapter 19 of the Standard Safety Analysis Report, Response to Severe Accident Policy Statement," SECY-91-309, October 1991.

PART II

Appendix A

Formal Questions and Answers on Sections 3.7 and 3.8

FIRST ROUND OF QUESTIONS ON THE LUNGMEN PSAR SECTIONS 3.7 AND 3.8

LUNGMEN PSAR SECTION 3.7

Section 3.7.1

1. Section 3.7.1.2: What method(s) will be used to develop the power spectral density functions corresponding to the site-specific design basis ground response spectra? What criteria will be used to evaluate the adequacy of the power spectral density functions?

2. Section 3.7.1.3: NUREG-0800 requires that correlation between stress levels and damping values be demonstrated. How will this requirement be met?

3. Section 3.7.1.3: What soil material damping values will be used? How will variations of soil material damping with strain be considered?

4. Section 3.7.1.3: Provide a technical justification for the damping values given in Table 3.7-1 for cable trays, conduits, and HVAC ductwork.

5. Section 3.7.1.4: NUREG-0800 requires additional information on the media supporting the Seismic Category I structures, including data on the soil profiles and soil properties. These data should be provided.

Section 3.7.2

1. Section 3.7.2.3.1: Provide the technical justification for acceptability of using the alternative approach (i.e., the dynamic substructuring concept) to account for dynamic interaction between the subsystem and the supporting structural system. What analyses have been or will be performed to substantiate the adequacy of this approach?

2. Section 3.7.2.3.1: Three-dimensional structural models are preferable compared to two-dimensional models. What analyses will be performed to verify the adequacy of any two-dimensional models used?

3. Section 3.7.2.3.3: Specify which sections of ASCE 4-86 are pertinent to the mathematical modeling and seismic analysis of the radwaste building.

4. Section 3.7.2.4: This section only notes that Computer Program SASSI will be used for the soil-structure interaction (SSI) analysis. Additional information to permit review of the SSI analysis with respect to NUREG-0800 requirements should be provided.

5. Section 3C.5: The discussion notes that various modifications to Computer Program SASSI have been performed. We have recently learned that use of the same nodes for the excavated soil model and interior foundation model, while implicitly permitted by the SASSI user's manual, can lead to erroneous results. In light of this, please verify that the Lungmen SASSI model is correctly developed. (See Appendix to this enclosure.)

6. Section 3.7.2.4: The PSAR notes that the Seismic Category I structures are to be founded on rock. Is there a shallow soil layer over rock? Where is the control point for input of earthquake ground motion?

7. Section 3.7.2.4: Describe how structure-soil-structure interaction between the independently founded structures has been considered.

8. Section 3.7.2.6: NUREG-0800 does not explicitly acknowledge acceptability of the 100-40-40 method for combining maximum responses due to the three ground motion components. Provide a technical justification for using this method.

9. Section 3.7.2.8: What are the definitions of Seismic Category IIA, IIB, and IIC structures?

10. Section 3.7.2.10: As stated in the PSAR, constant vertical static factors were not used since all Lungmen seismic Category I structures are subjected to a vertical dynamic analysis. How is the vertical flexibility of the floor slabs of the structures accounted for in the generation of vertical floor response spectra?

11. Describe how W_p and W_b terms in Equation (3.7-11) in Section 3.7.2.14 are calculated. Also, what is the acceptable factor of safety (i.e., ratio of E_0 to E_s) for stability against overturning moments for the OBE case?

Section 3.7.3

1. Review of Section 3.7.3 reveals only a limited number of references to OBE analysis. What types of analyses will be performed to account for differences in damping for OBE and SSE?

2. Section 3.7.3.1: Provide a technical justification for performing time-history analyses of a SRV blowdown without considering modeling uncertainties, i.e., by broadening the force time history by ±15%Δt.

3. Section 3.7.3.3.1.4: Provide a technical justification for the criteria on selection of relative anchor motions for decoupled branch pipe.

4. Section 3.7.3.3.4: Provide a technical justification for coefficients of friction used in the analysis of frame-type supports.

5. Section 3.7.3.8.8: Subsection II.9 of NUREG-0800, Section 3.7.3 requires that responses of multiply-supported equipment and components due to inertial effects and relative displacements be combined by the absolute sum, rather than SRSS. Provide a technical justification for this deviation.

6. Section 3.7.3.16: Provide damping values for the impulsive mode of tank contents.

Section 3.7.4

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

LUNGMEN PSAR SECTION 3.8

Section 3.8.1

1. Section 3.8.1.4.1.1 presents discussion on a three-dimensional finite element model of the containment structure and the reactor building. Symmetry of the containment structure and the reactor building about the centerline of the plant in a direction parallel to the fuel pool girders was considered such that only 180-degree of the structure was modeled for design analyses when subjected to the various load combinations. Explain how non-axisymmetric

loads are treated when using this 180-degree model for stress analyses for wind, earthquake, and hydrodynamic loads.

2. Explain the inconsistency between Note 6 of Table 3.8-1 and Section 3.8.1.4.1.1.1 where both the SRSS method and the 100-40-40 load factor method are specified for combining co-directional effects. Since the 100-40-40 combination method is not specified in RG 1.92, provide a technical justification for its use. (Also see Question 8 on Section 3.7.2.)

3. The "S" and "U" in Note 12 of Table 3.8-1 are termed as "required strength". The ASME Section III Division 2 code uses the term "allowable stresses". Explain why the terms used in the PSAR are not consistent with the ASME code.

4. The appendix to NUREG-0800 Section 3.8.1 states that the SRV loads should be treated as live loads in all load combinations with the exception of the combination that contains 1.5 P_a where a load factor of 1.25 should be applied to the appropriate SRV loads. The load combination No. 8 in Table 3.8-1 does not reflect this. Provide a technical justification for using a load factor of 1.0 instead of 1.25.

5. Are the concrete compressive allowable stresses given in Table 3.8-2 also applicable to primary stresses when subject to either membrane force or membrane force plus bending moment?

6. Are the v_{so} values given in Table 3.8-2 upper bound limits of tangential shear resistance that can be provided by the orthogonal reinforcement?

7. What are the values of the appropriate dynamic load factors that are used for the equivalent static loads to represent the dynamic loads resulting from a LOCA and SRV actuation (See Section 3.8.1.4.1.1.2)?

8. Describe how stresses and strains in the containment wall and liner plate at the major containment penetrations will be analyzed (Subsection 3.8.1.4.1.1.3.)

Section 3.8.2

1. In NUREG-1503, Appendix E the USNRC staff recommended using the criteria in Code Case N-284 to evaluate buckling of the ellipsoidal and torispherical drywell head under internal pressure. Provide a justification for not using such criterion in Section 3.8.2.4.1.4.

2. Define what is considered an "adequate factor of safety" for stability against compression buckling of metal components in Section 3.8.2.5.

Section 3.8.3

1. Section 3.8.3.4.2 states that the concrete placed between the outer and the inner steel cylindrical shells acts to distribute loads between the shells and to provide stability to the compression elements of the reactor pedestal. Provide discussions on how the load transfer will be carried out by the concrete. Is there reinforcing steel in the concrete between the concentric steel shells? What structural elements are used to tie the concrete with the steel shells to achieve composite action?

2. Provide discussions on how the base of the reactor pedestal is structurally connected to the containment basemat through the liner plate. How are the reactor pedestal loads transferred into the containment basemat?

3. Is there reinforcing steel in the concrete between the concentric steel shells of the reactor pedestal? What structural elements are used to tie the concrete with the steel shells to achieve composite action?

4. Describe how the reactor shield wall is connected to the reactor pedestal or to the diaphragm slab.

5. Questions on Table 3.8-5

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

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

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

   - Section 3.8.3.3.1 references the load combinations in Section 3.8.1.3 which includes the post-LOCA flooding. Explain why post-LOCA flooding is not one of the loads considered for the concrete internal structure.

6. For loads and load combinations considered for the steel internal structures, provide the following information:

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

   - Explain why post-LOCA flooding is not one of the load combinations considered?

7. Table 3.8-9 on the exceptions to ANSI/AISC N-690 is cited only in Section 3.8.4 of the PSAR. Why is it not cited in Section 3.8.3 where design of the steel internal structure is discussed?

8. Is the minimum compressive strength (27.6 MPa) of the concrete placed in the reactor pedestal and the reactor shield walls specified at 28 days or 90 days? (Subsections 3.8.3.6.2&3.) Are these normal weight or heavy height concrete?

9. Will bearing type or friction type A-325 high strength bolts be used for the drywell equipment and pipe support structure (DEPSS)?

Section 3.8.4

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

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

3. Provide in Section 3.8.4 the descriptions, applicable codes and standards, loads and load combinations, design and analysis procedures, and structural acceptance criteria for the refueling pool at the top of the containment structure.

4. Is the superstructure (the portion above the fuel pool) of the reactor building a seismic Category I structure? If it is, provide in Section 3.8.4 the descriptions, applicable codes and standards, loads and load combinations, design and analysis procedures, and structural acceptance criteria for this structure.

5. Explain why there are no discussions on the analysis and design and structural acceptance criteria for wind, typhoon, and typhoon-generated missiles in Section 3.8.4.

6. The third paragraph of Section 3.8.4.1.2 states that the Control Building is designed to provide missile and **tornado** protection. Is tornado a design basis load for Lungmen Units 1 and 2?

7. The first paragraph of Section 3.8.4.2.1 states that major portion of the R/B is not subjected to the abnormal and severe accident conditions associated with a containment. However, the R/B is structurally connected with the containment shell through the R/B floor slabs. Explain why the effects of the hydrodynamic loads from a SRV discharge or a LOCA in the containment are not considered in the R/B design.

8. Section 3.8.4.3.1.1: Explain why the SRV and LOCA loads on the RB are not included in the list of loads.

9. Section 3.8.4.3.1.1 - Loads and Notations. Explain why the effects of hydrostatic pressure due to design groundwater are not discussed. Also explain how seismic-induced dynamic soil pressure is derived.

10. It appears that the "H" term in the first load combination in Section 3.8.4.3.1.2 should be "H'" to represent total soil pressure (static + dynamic). Explain why the term should be "H."

11. Why is the following load combination not included in Section 3.8.4.3.1.3?

    $1.5S = D + L + To + Ro$ (NUREG-0800 Section 3.8.4 Paragraph II.3.c.(i)(a))

12. The floor design live load of 100 lbs/ft^2 for **all** floors of the R/B, the C/B, and the Auxiliary Fuel Building (Section 3.8.4.3.1.1 and 3.8.4.3.2) appears to be low. Provide basis for the use of this live load for all floors.

13. Explain why the allowable stress in Subsection 3.8.4.3.5.3 for the 4th load combination under working stress design should not be "1.33S" instead of "S."

14. What are the values of the factors of safety against sliding, overturning, and flotation for the reactor building, the control building, and the auxiliary fuel building (Subsection 3.8.4.4.1)?

15. Sections 3.8.4.4.2 and 3.8.4.4.3 cited "design by rule method" as one of the methods used for design and analysis procedures for seismic Category I HVAC, cable trays, conduits and their supports. Provide a technical justification for using the "design by rule method."

16. There are no equations provided in Subsection 3.8.4.4.4.2 for checking against sliding, overturning, and flotation for the radwaste building. Provide an explanation.

17. As discussed in Subsection 3.8.4.5.2, the control building is designed per the ultimate strength method in ACI 349. Then,

    - Why is the term "allowable stress" used? and
    - Why is the stress in the rebars limited to 0.9Fy?

18. Explain why Table 3.8-10 "Exceptions to ACI 349, Appendix B Steel Embedment" is not cited in the text of Section 3.8.

Section 3.8.5

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

APPENDIX TO

ENCLOSURE A

SASSI MODELING ERROR

FIRST ROUND OF QUESTIONS ON THE LUNGMEN FSAR
SASSI MODELING ERROR

SASSI Problem Statement

SASSI is a public domain computer program used to perform Soil-Structure Interaction (SSI) analysis for structures with rigid or flexible embedded foundations of arbitrary shape.

Based on recent discussions with several SASSI users, it is likely that many, if not most, of the SASSI models contain modeling errors in the foundation portion of the models. The modeling errors are a direct result of misleading instructions in the SASSI User's Manual. Depending on the type of foundation, soil properties, and type of results generated, the analysis results can have errors which range from negligible to very significant.

Background

The modeling error was discovered during the course of performing an SSI analysis of a nuclear power plant. Checks of dynamic equilibrium on interior foundation walls revealed that out-of-plane moments and shears in the walls were an order of magnitude higher than the moments and shears that could be produced by the peak absolute accelerations.

Considerable effort was expended performing studies and alternate analyses to understand the root cause of the problem. Eventually it was discovered that the problem stemmed from misleading instructions in the SASSI User's Manual on how to model the interior interaction nodes of the foundation.

The SASSI User's Manual states, "Two separate finite element models are constructed, one for the structure and the other for the excavated soil. The models share the same nodal points at/below the ground surface which are called the interaction nodes." A common practice, which was believed to be consistent with this statement, was to assign the same node number to coincident nodes of the interior foundation and excavated soil models, in the same way as is done for the exterior foundation nodes which are in contact with the soil. This practice, however, is not correct. The correct procedure is to assign different node numbers to the coincident nodes of the interior foundation and excavated soil models. That is, the interior of the excavated soil region should not be "attached" to any interior regions of the foundation.

Potential Magnitude of the Errors

As for the effect this modeling error can have on analysis results, limited comparisons from the SSI analysis show the following trends. Keep in mind that this is based on analysis of one particular plant.

Comparison between corrected and uncorrected SASSI models show the following:

- Horizontal response spectra in the super-structure and foundation increased by less than 1%. Vertical response spectra peaks increased by 10 to 25%. No frequency shifts occurred.

- For interior foundation walls, out-of-plane moment and shear decreased by a factor of 10. In the same walls in-plane forces decreased by approximately 30%.

- For exterior foundation walls, maximum out-of-plane moment and shear increased by as much as 65%. Maximum in-plane forces in the same walls increased by 10%.

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-020

PSAR Sections: 3.7.1.2

Question Date: March 6, 1998

PSAR Question:

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

Response:

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-020

PSAR Sections: 3.7.1.3

Question Date: March 6, 1998

PSAR Question:

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

Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-020

PSAR Sections: 3.7.1.3

Question Date: March 6, 1998

PSAR Question:

3. What soil material damping values will be used? How will variations of soil material damping with strain be considered?

Response:

3. PSAR Chapter 2.5 Tables 2.5-16 and 2.5-17 show the Strain-Dependent Shear Modulus reduction and damping factors for intact and weathered rocks and for crushed rock backfill that are used to meet this requirement.

   The following will be added to Sub Section 3.7.1.3 after the last sentence:

   Chapter 2.5 Tables 2.5-16 & 2.5-17 show the strain-dependent shear modulus reduction and damping factors for the supporting medium. Response of rock to dynamic loading is discussed in Sub Section 2.5.4.7.1 for seismic Category I structures.

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-020

PSAR Sections: 3.7.1.3

Question Date: March 6, 1998

PSAR Question:

4. Provide a technical justification for the damping values given in Table 3.7-1 for cable trays, conduits, and HVAC ductwork.

PSAR Response:

4. The damping values for cable trays, conduits, and HVAC ductwork are in compliance with RG 1.61 values for bolted or welded structure as appropriate.

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

A-14

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-020

PSAR Sections: 3.7.1.4

Question Date: March 6, 1998

PSAR Question:

5. NUREG-0800 requires additional information on the media supporting the seismic Category I structures, including data on the soil profiles and soil properties. These data should be provided.

Response:

5. PSAR Subsection 2.5.4.2.4 lists the Engineering Properties of the foundation materials to be used on Lungmen NPS project. The existing soil profiles at the Plant site are discussed in PSAR Subsection 2.5.4.2.3.

The following will be added to PSAR Sub Section 3.7.1.4 at the end of last paragraph:

The existing soil profile at the plant site are discussed in Sub Section 2.5.4.2.3. Sub Section 2.5.4.2.4 lists the engineering properties of the foundation materials for use on Lungmen NPS project.

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-021

PSAR Sections: 3.7.2.3.1

Question Date: March 6, 1998

PSAR Question:

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

PSAR Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-021

PSAR Sections: 3.7.2.3.1

Question Date: March 6, 1998

PSAR Question:

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

PSAR Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-021

PSAR Sections: 3.7.2.3.3

Question Date: March 6, 1998

PSAR Question:

3. Specify which sections of ASCE 4-86 are pertinent to the mathematical modeling and seismic analysis of the radwaste building.

Response:

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

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-021

PSAR Sections: 3.7.2.4

Question Date: March 6, 1998

PSAR Question:

4. This section only notes that Computer Program SASSI will be used for the soil-structure interaction (SSI) analysis. Additional information to permit review of the SSI analysis with respect to NUREG-0800 requirements should be provided.

PSAR Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-021

PSAR Sections: 3C.5

Question Date: March 6, 1998

PSAR Question:

5. The discussion notes that various modifications to Computer Program SASSI have been performed. We have recently learned that use of the same nodes for the excavated soil model and interior foundation model, while implicitly permitted by the SASSI user's manual, can lead to erroneous results. In light of this, please verify that the Lungmen SASSI model is correctly developed. (See Appendix to this enclosure)

PSAR Response:

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-021

PSAR Sections: 3.7.2.4

Question Date: March 6, 1998

PSAR Question:

6. The PSAR notes that the seismic Category I structures are to be founded on rock. Is there a shallow soil layer over rock? Where is the control point for input of earthquake ground motion?

PSAR Response:

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-021

PSAR Sections: 3.7.2.4

Question Date: March 6, 1998

PSAR Question:

7. Describe how structure-soil-structure interaction between the independently founded structures has been considered.

PSAR Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-021

PSAR Sections: 3.7.2.6

Question Date: March 6, 1998

PSAR Question:

8. NUREG-0800 does not explicitly acknowledge acceptability of the 100-40-40 method for combining maximum responses due to the three ground motion components. Provide a technical justification for using this method.

Response:

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-021

PSAR Sections: 3.7.2.8

Question Date: March 6, 1998

PSAR Question:

9. What are the definitions of Seismic Category IIA, IIB, and IIC structures?

Response:

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-021

PSAR Sections: 3.7.2.10

Question Date: March 6, 1998

PSAR Question:

10. As stated in the PSAR, constant vertical static factors were not used since all Lungmen seismic Category I structures are subjected to a vertical dynamic analysis. How is the vertical flexibility of the floor slabs of the structures accounted for in the generation of vertical floor response spectra?

PSAR Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-021

PSAR Sections:

Question Date: March 6, 1998

PSAR Question:

11. Describe how W_p and W_b terms in Equation (3.7-11) in Section 3.7.2.14 are calculated. Also, what is the acceptable factor of safety (i.e., ratio of E_0 to E_1) for stability against overturning moments for the OBE case?

PSAR Response:

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

$$W_p = \int_0^d p_z C z \tan\theta dz$$

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

The work done by the buoyancy force is given by,

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

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-022

PSAR Sections:

Question Date: March 6, 1998

PSAR Question:

1. Review of Section 3.7.3 reveals only a limited number of references to OBE analyses. What types of analyses will be performed to account for differences in damping for OBE and SSE?

Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-022

PSAR Sections: 3.7.3.1

Question Date: March 6, 1998

PSAR Question:

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

Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-022

PSAR Sections: 3.7.3.3.1.4

Question Date: March 6, 1998

PSAR Question:

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

PSAR Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-022

PSAR Sections: 3.7.3.3.4

Question Date: March 6, 1998

PSAR Question:

4. Provide a technical justification for coefficients of friction used in the analysis of frame-type supports.

PSAR Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-022

PSAR Sections: 3.7.3.8.8

Question Date: March 6, 1998

PSAR Question:

5. Subsection II.9 of NUREG-0800, Section 3.7.3 requires that responses of multiply-supported equipment and components due to inertial effects and relative displacements be combined by the absolute sum, rather than SRSS. Provide a technical justification for this deviation.

PSAR Response:

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-022

PSAR Sections: 3.7.3.16

Question Date: March 6, 1998

PSAR Question:

6. Provide damping values for the impulsive mode of tank contents.

PSAR Response:

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-023

PSAR Sections:

Question Date: March 6, 1998

PSAR Question:

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

Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-024

PSAR Sections:

Question Date: March 6, 1998

PSAR Question:

1. Section 3.8.1.4.1.1 presents a discussion on a three-dimensional finite element model of the containment structure and the reactor building. Symmetry of the containment structure and the reactor building about the centerline of the plant in a direction parallel to the fuel pool girders was considered such that only 180-degree of the structure was modeled for design analyses when subjected to the various load combinations. Explain how non-axisymmetric loads are treated when using the 180-degree model for stress analyses for wind, earthquake, and hydrodynamic loads.

2. Explain the inconsistency between Note 6 of Table 3.8-1 and Section 3.8.1.4.1.1.1 where both the SRSS method and the 100-40-40 load factor method are specified for combining co-directional effects. Since the 100-40-40 combination method is not specified in RG 1.92, provide a technical justification for its use. (Also see Question 8 on Section 3.7.2.)

PSAR Response:

1. For non-axisymmetric loads the boundary displacements for nodes at the plane of symmetry follow the anti-symmetric conditions.

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-024

PSAR Sections: 3.8.1

Question Date: March 6, 1998

PSAR Question:

3. The "S" and "U" in Note 12 of Table 3.8-1 are termed as "required strength". The ASME Section III Division 2 code uses the term "allowable stresses". Explain why the terms used in the PSAR are not consistent with the ASME code.

PSAR Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-025

PSAR Sections:

Question Date: March 6, 1998

PSAR Question:

1. The appendix to NUREG-0800 Section 3.8.1 states that the SRV loads should be treated as live loads in all load combinations with the exception of the combination that contains 1.5 P_a where a load factor of 1.25 should be applied to the appropriate SRV loads. The load combination No. 8 in Table 3.8-1 does not reflect this. Provide a technical justification for using a load factor of 1.0 instead of 1.25.

PSAR Response:

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

   The 1.25 load factor for SRV as specified in Appendix to NUREG-0800, Section 3.8.1 is a conservative value due to lack of full understanding of the SRV loading in 1970's. Since then, knowledge gained from plant tests has confirmed that the inherent margins in the SRV load definition are sufficiently conservative and further increase by using 1.25 load factor is not warranted. Thus, a more realistic factor of 1.0 as specified in the 1989 ASME Code Section III, Division 2, Sub Section CC is technically adequate.

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-025

PSAR Sections: 3.8.1

Question Date: March 6, 1998

PSAR Question:

2. Are the concrete compressive allowable stresses given in Table 3.8-2 also applicable to primary stresses when subject to either membrane force or membrane force plus bending moment?

PSAR Response:

2. No. The allowable concrete compressive stresses are the maximum allowable corresponding to membrane force + bending moment in accordance with Tables CC-3421-1 and CC-3431-1 of ASME B&PV Code, Section III, Division 2, Subsection CC.

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-025

PSAR Sections: 3.8.1

Question Date: March 6, 1998

PSAR Question:

3. Are the v_{so} values given in Table 3.8-2 upper bound limits of tangential shear resistance that can be provided by the orthogonal reinforcement?

Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-025

PSAR Sections:

Question Date: March 6, 1998

PSAR Question:

4. What are the values of the appropriate dynamic load factors that are used for the equivalent static loads to represent the dynamic loads resulting from a LOCA and SRV actuation (see Section 3.8.1.4.1.1.2)?

Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-025

PSAR Sections: 3.8.1

Question Date: March 6, 1998

PSAR Question:

5. Describe how stress and strains in the containment wall and liner plate at the major containment penetrations will be analyzed (Subsection 3.8.1.4.1.1.3).

PSAR Response:

5. PSAR Subsection 3.8.1.4.1.1.3 documents the major penetrations in the containment wall. These penetrations are modeled in the global finite element model with a finer mesh. The model also includes the liner plate. The liner strains are taken directly from this global model analysis results. The analysis of stresses in the concrete surrounding these openings and the reinforcing bars therein, is performed using computer code SSDP-2D as discussed in PSAR Appendix Section 3C.3.

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-026

PSAR Sections: 3.8.2

Question Date: March 6, 1998

PSAR Question:

1. In NUREG-1503, Appendix E the USNRC staff recommended using the criteria in Code Case N-284 to evaluate buckling of the ellipsoidal and torispherical drywell head under internal pressure. Provide a justification for not using such criterion in Section 3.8.2.4.1.4.

PSAR Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-026

PSAR Sections: 3.8.2

Question Date: March 6, 1998

PSAR Question:

2. Define what is considered an "adequate factor of safety" for stability against compression buckling of metal components in Section 3.8.2.5.

PSAR Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-027

PSAR Sections: 3.8.3

Question Date: March 6, 1998

PSAR Question:

1. Section 3.8.3.4.2 states that the concrete placed between the outer and the inner steel cylindrical shells acts to distribute loads between the shells and to provide stability to the compression elements of the reactor pedestal. Provide discussions on how the load transfer will be carried out by the concrete. Is there reinforcing steel in the concrete between the concentric steel shells? What structural elements are used to tie the concrete with the steel shells to achieve composite action?

PSAR Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-027

PSAR Sections: 3.8.3

Question Date: March 6, 1998

PSAR Question:

2. Provide discussions on how the base of the reactor pedestal is structurally connected to the containment basemat through the liner plate. How are the reactor pedestal loads transferred into the containment basemat?

PSAR Response:

2. The Reactor Pedestal cylindrical wall shell plates are welded to the base plate at their lower edges. The base plate is welded to embedded stiffener plates below elevation -8200 mm. In addition, the Pedestal is anchored by anchor bolts embedded in concrete basemat.

   The transfer of load from Pedestal takes place through (1) direct bearing on base plate, (2) Shear transfer through embedded stiffener plates, and (3) by anchor bolts tension.

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-027

PSAR Sections: 3.8.3

Question Date: March 6, 1998

PSAR Question:

3. Is there reinforcing steel in the concrete between the concentric steel shells of the reactor pedestal? What structural elements are used to tie the concrete with the steel shells to achieve composite action?

PSAR Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-027

PSAR Sections:

Question Date: March 6, 1998

PSAR Question:

4. Describe how the reactor shield wall is connected to the reactor pedestal or to the diaphragm slab.

5. Questions on Table 3.8-5:

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

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

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

   (4) Section 3.8.3.3.1 references the load combinations in Section 3.8.1.3 which includes the post-LOCA flooding. Explain why post-LOCA flooding is not one of the loads considered for the concrete internal structure.

PSAR Response:

4. The Reactor Shield Wall is connected to the Pedestal shell plates by full penetration welds.

   The reactor shield wall vertical plates are sitting directly on top of the pedestal inner most vertical plates and connected to them by full penetration welds. The diaphragm floor slab, between the elevations of 11100 and 12300, is connected to the outermost vertical plate of the pedestal and is sitting on the pedestal. Thus, the reactor shield wall is not directly connected to diaphragm floor slab.

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

5. (1) The dynamic load factors for Pa, Ta, and R_a are 1.0. For Y, the factors will depend on load distribution and the frequency of forcing functions. For hydrodynamic loads please refer to response to question 4 of Track No. 03-025

   (2) Please refer to response to question 8 of Track No. 03-021.

(3) Please refer to response to question 3 of Track No. 03-024

(4) Post-LOCA flooding load case is required for the design of containment structure only. It is not required for the design of containment internal structures as per the design code ACI 349 for concrete and AISC N690 for steel, nor is the SRP 3.8.3 requirement.

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-027

PSAR Sections: 3.8.3

Question Date: March 6, 1998

PSAR Question:

6. For loads and load combinations considered for the steel internal structures, provide the following information:

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

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

PSAR Response:

6. (1) The acceptance criteria for bolts is in accordance with ANSI/AISC N-690 as indicated in Subsection 3.8.3.2.

(2) Please see response to Question 5 above.

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-027

PSAR Sections: 3.8.3

Question Date: March 6, 1998

PSAR Question:

7. Table 3.8-9 on the exceptions to ANSI/AISC N-690 is cited only in Section 3.8.4 of the PSAR. Why is it not cited in Section 3.8.3 where design of the steel internal structures is discussed?

PSAR Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-027

PSAR Sections: 3.8.3

Question Date: March 6, 1998

PSAR Question:

8. Is the minimum compressive strength (27.6 MPa) of the concrete placed in the reactor pedestal and the reactor shield walls specified at 28 days or 90 days? (Subsections 3.8.3.6.2 and 3.8.3.6.3.) Are these normal weight or heavy height concrete?

PSAR Response:

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

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

A-50.

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-027

PSAR Sections: 3.8.3

Question Date: March 6, 1998

PSAR Question:

9. Will bearing type or friction type A-325 high strength bolts be used for the drywell equipment and pipe support structure (DEPSS)?

PSAR Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-028

PSAR Sections: 3.8.4

Question Date: March 6, 1998

PSAR Question:

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

Response:

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

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track No.: 03-028

PSAR Sections: 3.8.4

Question Date: MAR. 4, 1998

PSAR Question:

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

PSAR Response:

Safety related Seismic Category I, piping systems and their supports are designed in accordance with ASME III.

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-028

PSAR Sections:

Question Date: March 6, 1998

PSAR Question:

3. Provide in Section 3.8.4 the descriptions, applicable codes and standards, loads and load combinations, design and analysis procedures, and structural acceptance criteria for the refueling pool at the top of the containment structure.
4. Is the superstructure (the portion above the fuel pool) of the reactor building a seismic Category I structure? If it is, provide in Sections 3.8.4 the descriptions, applicable codes and standards, loads and load combinations, design and analysis procedures, and structural acceptance criteria for this structure.
5. Explain why there are no discussions on the analysis and design and structural acceptance criteria for wind, typhoon, and typhoon-generated missiles in Section 3.8.4.
17. As discussed in Subsection 3.8.4.5.2, the control building is designed per the ultimate strength method in ACI 349. Then,

    (1) Why is the term "allowable stress" used? and

    (2) Why is the stress in the rebars limited to 0.9Fy?

PSAR Response:

3. The loads, loading combinations, and acceptance criteria applicable to Reactor Building, discussed in Section 3.8.4, are also applicable to refueling pool. Subsection 3.8.4.1.1 refers to refueling pool girders.

The finite element model for the integrated Reactor Building and the RCCV discussed in Section 3.8.1.4.1.1 includes the refueling pool structure. The calculated forces and moments for various loads will be used for design evaluation.

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

4. As indicated in Subsection 3.8.4.1.1 the superstructure of the Reactor Building consists of steel beams and columns and is a seismic Category I structure. The loads, loading combinations, and acceptance criteria applicable to Reactor Building, defined in Section 3.8.4, are also applicable to its superstructure.

The finite element model for the integrated Reactor Building and the RCCV discussed in Section 3.8.1.4.1.1 includes the superstructure. The calculated forces and moments for various loads will be used for design evaluation of the Reactor Building superstructure.

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

A-54

RESPONSX

RESPONSES TO MPR's PSAR QUESTIONS

5. The wind load,W and typhoon load, W_t and typhoon generated missiles are defined in Subsection 3.8.4.3.1.1 and used in load combinations in Subsections 3.8.4.3.1.2 and 3.8.4.3.1.3. No separate structural acceptance criteria is required for wind loads alone. The wind load is discussed in PSAR Section 3.3.1. The typhoon load and the typhoon generated missiles are described in PSAR Section 3.3.2. The analysis and design procedures for wind and typhoon loads are based on ANSI/ASCE 7-90 as cited in PSAR Sections 3.3.1 and 3.3.2.

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

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

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-028

PSAR Sections: 3.8.4

Question Date: March 6, 1998

PSAR Question:

6. The third paragraph of Section 3.8.4.1.2 states that the Control Building is designed to provide missile and tornado protection. Is tornado a design basis load for Lungmen Units 1 and 2?

Response:

6. There is no tornado design requirements for the Lungmen NPS.

   This typographical error in PSAR Section 3.8.4.1.2 will be corrected and "tornado" will be changed to "typhoon."

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-028

PSAR Sections: 3.8.4

Question Date: March 6, 1998

PSAR Question:

7. The first paragraph of Section 3.8.4.2.1 states that major portion of the R/B is not subjected to the abnormal and severe accident conditions associated with a containment. However, the R/B is structurally connected with the containment shell through the R/B floor slabs. Explain why the effects of the hydrodynamic loads from a SRV discharge or a LOCA in the containment are not considered in the R/B design.

PSAR Response:

7. The effects of the hydrodynamic loads, from a SRV discharge or a LOCA in the containment, on the R/B structure are accounted for using an integrated R/B and RCCV analysis model.

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-028

PSAR Sections: 3.8.4.3.1.1

Question Date: March 6, 1998

PSAR Question:

8. Explain why the SRV and LOCA loads on the RB are not included in the list of loads.

PSAR Response:

8. Please see response to Question 7 above.

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-028

PSAR Sections: 3.8.4.3.1.1

Question Date: March 6, 1998

PSAR Question:

9. Loads and Notations. Explain why the effects of hydrostatic pressure due to design groundwater are not discussed. Also explain how seismic-induced dynamic soil pressure is derived.

PSAR Response:

9. Hydrostatic pressure is treated as part of static soil pressure. Seismic-induced dynamic soil pressures are calculated from soil-structure interaction analysis.

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-028

PSAR Sections: 3.8.4.3.1.2

Question Date: March 6, 1998

PSAR Question:

10. It appears that the "H" term in the first load combination in Section 3.8.4.3.1.2 should be "H" to represent total soil pressure (static + dynamic). Explain why the term should be "H."

PSAR Response:

10. GE will correct the typographical error and will change H to H', where H' represents total soil pressure, either due to OBE or SSE event.

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-028

PSAR Sections: 3.8.4.3.1.3

Question Date: March 6, 1998

PSAR Question:

11. Why is the following load combination not included in Section 3.8.4.3.1.3?

    1.5S = D + L + To + Ro (NUREG-0800 Section 3.8.4 Paragraph II.3.c.(i)(a))

PSAR Response:

11. The two load cases included in Subsection 3.8.4.3.1.3 are more critical compared to the above load case since each one of them includes the above loads and include OBE in one load case and wind load in the other load case.

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

A-61

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-028

PSAR Sections: 3.8.4

Question Date: March 6, 1998

PSAR Question:

12. The floor design live load of 100 lbs/ft^2 for all floors of the R/B, the C/B, and the Auxiliary Fuel Building (Section 3.8.4.3.1.1 and 3.8.4.3.2) appears to be low. Provide basis for the use of this live load for all floors.

PSAR Response:

12. Actual calculated Live Loads in Reactor and Control Buildings, for normal operations, are determined to be less than 100 lbs/ft^2.

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

A-62

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-028

PSAR Sections: 3.8

Question Date: March 6, 1998

PSAR Question:

13. Explain why the allowable stress in Subsection 3.8.4.3.5.3 for the fourth load combination under working stress design should not be "1.33S" instead of " S."

Response:

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-028

PSAR Sections:

Question Date: March 6, 1998

PSAR Question:

14. What are the values of the factors of safety against sliding, overturning, and flotation for the reactor building, the control building, and the auxiliary fuel building (Subsection 3.8.4.4.1)?

Response:

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

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-028

PSAR Sections: 3.8.4.4.2 and 3.8.4.4.3

Question Date: March 6, 1998

PSAR Question:

15. Sections 3.8.4.4.2 and 3.8.4.4.3 cited "design by rule method" as one of the methods used for design and analysis procedures for seismic Category I HVAC, cable trays, conduits and their supports. Provide a technical justification for using the "design by rule method."

PSAR Response:

15. Reference to "design by rule method" will be deleted from these sections.

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

A-65

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 03-028

PSAR Sections: 3.8

Question Date: March 6, 1998

PSAR Question:

16. There are no equations provided in Subsection 3.8.4.4.4.2 for checking against sliding, overturning, and flotation for the radwaste building. Provide an explanation.

Response:

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

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

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-028

PSAR Sections:

Question Date: March 6, 1998

PSAR Question:

3. Provide in Section 3.8.4 the descriptions, applicable codes and standards, loads and load combinations, design and analysis procedures, and structural acceptance criteria for the refueling pool at the top of the containment structure.
4. Is the superstructure (the portion above the fuel pool) of the reactor building a seismic Category I structure? If it is, provide in Sections 3.8.4 the descriptions, applicable codes and standards, loads and load combinations, design and analysis procedures, and structural acceptance criteria for this structure.
5. Explain why there are no discussions on the analysis and design and structural acceptance criteria for wind, typhoon, and typhoon-generated missiles in Section 3.8.4.

→ 17. As discussed in Subsection 3.8.4.5.2, the control building is designed per the ultimate strength method in ACI 349. Then,

(1) Why is the term "allowable stress" used? and

(2) Why is the stress in the rebars limited to 0.9Fy?

PSAR Response:

3. The loads, loading combinations, and acceptance criteria applicable to Reactor Building, discussed in Section 3.8.4, are also applicable to refueling pool. Subsection 3.8.4.1.1 refers to refueling pool girders.

The finite element model for the integrated Reactor Building and the RCCV discussed in Section 3.8.1.4.1.1 includes the refueling pool structure. The calculated forces and moments for various loads will be used for design evaluation.

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

4. As indicated in Subsection 3.8.4.1.1 the superstructure of the Reactor Building consists of steel beams and columns and is a seismic Category I structure. The loads, loading combinations, and acceptance criteria applicable to Reactor Building, defined in Section 3.8.4, are also applicable to its superstructure.

The finite element model for the integrated Reactor Building and the RCCV discussed in Section 3.8.1.4.1.1 includes the superstructure. The calculated forces and moments for various loads will be used for design evaluation of the Reactor Building superstructure.

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

5. The wind load,W and typhoon load, W_t and typhoon generated missiles are defined in Subsection 3.8.4.3.1.1 and used in load combinations in Subsections 3.8.4.3.1.2 and 3.8.4.3.1.3. No separate structural acceptance criteria is required for wind loads alone. The wind load is discussed in PSAR Section 3.3.1. The typhoon load and the typhoon generated missiles are described in PSAR Section 3.3.2. The analysis and design procedures for wind and typhoon loads are based on ANSI/ASCE 7-90 as cited in PSAR Sections 3.3.1 and 3.3.2.

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

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

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-028

PSAR Sections: 3.8.4

Question Date: March 6, 1998

PSAR Question:

18. Explain why Table 3.8-10 "Exceptions to ACI 349, Appendix B Steel Embedment" is not cited in the text of Section 3.8.

PSAR Response:

18. Section 3.8.3.2 states that Diaphragm Floor slab design shall be in accordance with Reference Number 13. Reference 13 of Table 3.8.4 lists the ACI 349 Code with the exceptions noted in Table 3.8-10.

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 03-029

PSAR Sections: 3.8.5.4

Question Date: March 6, 1998

PSAR Question:

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

PSAR Response:

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

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

Appendix B

Formal Questions and Answers on Chapter 7

FIRST ROUND OF QUESTIONS ON THE LUNGMEN PSAR
CHAPTER 7

General

1. 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.

2. 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.

3. 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.

Section 7.1

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 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 the 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, 1.152, and 1.153; and HCIB 11, 12, 14, 17, 18, and 21 for the 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 the 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.

Section 7.2

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

Section 7.4

1. 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.

RESPONSES TO MPR'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 MPR's PSAR QUESTIONS

Track Number: 07-013

PSAR Sections: General

Question Date: March 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 MPR'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 MPR'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 MPR'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 MPR's PSAR QUESTIONS

Track Number: 07-017

PSAR Sections: 7.4

Question Date: March 6, 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 will be made to the PSAR as a result of the response to this question.

Appendix C

Formal Questions and Answers on Appendix 9A

QUESTIONS ON LUNGMEN NPS PSAR APPENDIX 9A

QUESTION 9A.1: FIRE HAZARDS REVIEW OF AREAS 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 safely 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.

5/20/98 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.

FOLLOW UP QUESTION

Response is considered acceptable. No follow up question.

QUESTION 9A.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.

5/20/98 PSAR Response:

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.

FOLLOW UP QUESTION

Response is considered acceptable. No follow up question.

QUESTION 9A.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.

5/20/98 PSAR Response:

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.

The Lungman Site Fire Protection Program will be in place and will address emergency action in case of fires on the construction site.

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.

FOLLOW UP QUESTION

Response is considered acceptable. No follow up question.

QUESTION 9A.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 from 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.

4/3/98 PSAR Response:

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.

FOLLOW UP QUESTION

With regard to smoke from external fires migrating into buildings via the HVAC systems, one system of particular concern is the RB secondary containment HVAC

system. This system provides ventilation air to rooms for all three safety divisions. Smoke from an external fire may potentially be distributed to safety-related areas for all three divisions, resulting in alarms from smoke detectors in each of these areas. This could result in operator confusion. Also, to remove smoke that migrated into each divisional area, operation in the smoke removal mode may not be possible until the smoke from the external fire clears from the ventilation system inlets. Will this system contain combination, operable smoke-fire dampers and smoke detectors in the supply duct? Describe other design features that minimize the impact of this common mode scenario for the RB secondary containment HVAC system.

QUESTION 9A.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 bypasses 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.

4/3/98 PSAR Response:

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.

FOLLOW UP QUESTION

Response is considered acceptable. No follow up question.

QUESTION 9A.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?

4/3/98 PSAR Response:

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.

FOLLOW UP QUESTION

Response is considered acceptable. No follow up question.

QUESTION 9A.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.

4/3/98 PSAR Response:

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.

FOLLOW UP QUESTION

Response is considered acceptable. No follow up question.

QUESTION 9A.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.

5/20/98 PSAR Response:

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.

FOLLOWUP QUESTION

Response is considered acceptable. No follow up question.

QUESTION 9A.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

than 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's 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 propagating 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.

4/3/98 PSAR Response:

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.

FOLLOW UP QUESTION

Response is considered acceptable. No follow up question.

QUESTION 9A.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 and 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 formal FHA.

4/3/98 PSAR Response:

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

FOLLOW UP QUESTION

The last sentence of the response is not complete.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track No. 09-030

PSAR Sections: 9.5.1, 9A, 9B

Question Date: March 2, 1998

PSAR Question:

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 No. 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 No. 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:

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.
The Lungman Site Fire Protection Program will be in place and will address emergency action in case of fires on the construction site.
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 MPR'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 MPR'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 MPR'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 MPR'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 MPR'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 MPR'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

Appendix D

Formal Questions and Answers on Chapter 19

FIRST ROUND REVIEW QUESTIONS ON LUNGMEN PSAR CHAPTER 19 SEVERE ACCIDENT ANALYSIS

SECTION 19.0 - SEVERE ACCIDENT ANALYSIS

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

SECTION 19.1 - INTRODUCTION

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

SECTION 19.2 - OVERVIEW

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

2. On page 19.2-3, it is indicated that only one RBCW/RBSW ESF division is capable of removing heat loads associated with operation of ECCS pumps and only one division is capable of removing suppression pool heat during LOCA. Please explain how the independence and redundancy requirements are satisfied for the heat removal function provided by the RBCW/RBSW.

SECTION 19.3 - ASSESSMENT OF SEVERE ACCIDENT PREVENTION

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

SECTION 19.3.1 - SEVERE ACCIDENT PREVENTIVE FEATURES

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

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

    - Frequencies of accident initiators
    - Conditional likelihood of safety function failure given occurrence of an accident initiator. This can be further divided into:
        - Conditional failure likelihood of the scram function given occurrence of an accident initiator.
        - Conditional failure likelihood of the pressure relief function given occurrence of an accident initiator (with success and failure of the scram function separately).

- Conditional likelihood of a transient-induced LOCA given occurrence of an accident initiator.

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

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

- Conditional failure likelihood of the low pressure injection function given occurrence of an accident initiator, successful scram function, failure of high pressure injection, and successful vessel depressurization.

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

- Conditional likelihood of core damage given occurrence of an accident initiator.

- Conditional likelihood of core damage given a transient-induced LOCA.

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

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

SECTION 19.3.1.1 - ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

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

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

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

SECTION 19.3.1.2 - STATION BLACKOUT (SBO)

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

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

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

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

SECTION 19.3.1.3 - FIRE PROTECTION

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

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

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

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

SECTION 19.3.1.4 - INTERSYSTEM LOSS OF COOLANT ACCIDENT (ISLOCA)

1. Resolution of the ISLOCA issue requires that low pressure piping systems interfacing with the reactor coolant pressure boundary (RCPB) be designed to withstand reactor pressure to the extent practicable. For some low pressure systems attached to the RCPB, it may not be practicable or necessary to provide a higher system ultimate pressure capability for the entire low pressure connected system. The systems interfacing with the

RCPB that have not been designed to withstand the full reactor pressure should include (1) the capability for leak testing the pressure isolation valves, (2) indication in the control room of valve position when isolation valve operators are deenergized, and (3) high pressure alarm to warn control room operators when rising reactor pressure approaches the design pressure of attached low pressure systems or when both isolation valves are not closed. Please provide information to show which of these additional protection features will be included in the Lungmen NPS design for each of the applicable systems that have been determined to interface directly or indirectly with the RCPB.

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

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

SECTION 19.4.1 – OVERVIEW OF THE CONTAINMENT DESIGN

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

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

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

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

SECTION 19.4.2 - SEVERE ACCIDENT PROGRESSION

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

SECTION 19.4.3 – SEVERE ACCIDENT MITIGATIVE FEATURES

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

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

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

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

5. The decomposition event tree concept is introduced in Section19.4.3.2.2.1 with essentially no explanation. Has the reader been introduced to this concept earlier in the PSAR? There should be a brief introductory explanation as to the purpose of these trees and how they relate to traditional containment event trees.

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

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

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

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

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

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

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

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

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

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

16. In the second sentence of the fourth paragraph of Section 19.4.3.6.3.2, "impulse loading" or "impulse pressure" is missing. It is also noted that "the pressure experienced by the pedestal wall will be reduced because the shock wave has to pass through some amount of water before it impinges on the

wall." Won't the existence of a free surface also diminish the shock wave as it passes through the water?

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

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

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

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

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

RECOMMENDATIONS/TYPOS

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

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

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

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

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

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

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

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

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

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

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

References:

1. Pilch, M.M., et al, Resolution of the Direct Containment Heating Issue for All Westinghouse Plants With Large Dry Containment or subatmospheric Containments," NUREG/CR-6338, February 1996.

SECOND ROUND OF QUESTIONS ON THE LUNGMEN PSAR CHAPTER 19 SEVERE ACCIDENT ANALYSIS

Section 19.1 – Introduction

1. The response to First Round Question 1 on PSAR Section 19.1 (Track No. 19-021) indicates that an SRP addressing severe accident analysis, Chapter 19, specifically developed for Lungmen NPS is attached to the response to the first round questions. However, this attachment of the "SRP Chapter 19" cannot be found in the response to the first round questions. The response to Question 1 on PSAR Section 19.1 also refers to the response to Question 19-016 for the basis of PSAR Chapter 19 contents and organization. This "response to Question 19-016" is also not included in the response to the first round questions. Please provide the "SRP Chapter 19" used in the preparation of Lungmen PSAR and "response to Question 19-016".

   The USNRC Final Safety Evaluation Report (FSER) for the Standard ABWR was prepared for the certification of a generic design without considering a specific site. The contents and key severe accident issues will not be necessarily the same for the application of a construction permit at a specific site. For instance, regarding the deterministic assessment of severe accident prevention, plant features intended for the response to and/or mitigation of both internal and external hazards should be summarized. The need to address the external hazards is especially important at sites with significant, known natural hazards.

   The content of PSAR should include a preliminary analysis and evaluation of the design and performance of the facility with the objective of assessing the risk to public health and safety (10CFR 50.34) including determination of

   - The margins of safety during normal operations and transient conditions, and
   - The adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

   From the standpoint of public health and safety, the considerations of accidents should not be limited to those initiated from the internal hazards only.

   Even though some of the design information is already provided in details in other sections of the PSAR, information relevant to severe accident prevention and mitigation should be summarized both deterministically and probabilistically in an integrated manner in Chapter 19. For each of the safety functions, conformance to the defense-in-depth philosophy as well as redundancy, independence, and diversity requirements should be discussed. Probabilistic measures of the plant's ability to prevent and mitigate severe accidents (e.g., reliability of safety functions and conditional likelihood of core damage for various accident initiators) should also be provided in Chapter 19.

   In addition, differences in design and operations between the Lungmen NPS and Standard ABWR that are relevant to severe accident prevention and mitigation should

be summarized. The impact of these differences on Lungmen's ability to prevent and mitigate severe accidents should also be addressed in Chapter 19.

Will the Lungmen PSAR be updated to include in Chapter 19 the additional items discussed in the preceding?

Section 19.3 – **Assessment of Severe Accident Prevention**

1. Is the swing EDG the only difference between Lungmen NPS and Standard ABWR that is relevant to severe accident prevention? If not, please provide a complete list of the differences in design and operations between the Lungmen NPS and Standard ABWR that are applicable to severe accident prevention. Furthermore, please also provide an evaluation of the effects of these differences on severe accident prevention and mitigation.

Section 19.3.1 – **Severe Accident Preventive Features**

1. The response to First Round Question 1 on Section 19.3.1 (Track No.19-024) indicates that Unresolved Safety Issue (USI) A-45 is resolved for the Lungmen NPS. It is true that A-45 was considered resolved for plants that have completed an Individual Plant Examination (IPE) or a risk assessment and have demonstrated that there are no vulnerabilities related to shutdown decay heat removal (DHR). Contributions to core damage and severe release frequencies due to failure of the DHR function must be acceptably low. Major contributors to core damage or severe release due to DHR function failure should be explicitly identified. In addition, cost-effective corrective actions for these major contributors, if any, should have been evaluated. While the total core damage and severe release frequencies at the Lungmen NPS are less than the USNRC's safety goal guidance, it would still be a better documentation to provide a separate section using results from the Lungmen PRA to demonstrate that there are no vulnerabilities related to DHR function failure as was done in most of the U.S. IPE programs (i.e., no individual DHR failure contributors dominate the core damage or severe release risk). Will the Lungmen PSAR be updated to include the above information in a separate section?

2. The response to First Round Question 2 on Section 19.3.1 (Track No. 19-024) refer to Appendix A and Attachments AA and AB of Appendix A. However, most of the probabilistic parameters listed in this question (e.g., likelihood of safety function failure) cannot be found in Appendix A and its attachments. This question remains to be an open issue. Will the Lungmen PSAR be updated to present these probabilistic parameters in Chapter 19?

3. The response to First Round Question 3 on Section 19.3.1 (Track No. 19-024) indicates that external events, including earthquake and typhoon are outside the scope of Chapter 19 according to the "Lungmen Chapter 19 SRP". The document referred to in this response, i.e., Lungmen Chapter 19 SRP, is not a "Standard Review Plan" issued by any regulatory agency. This question remains to be an open issue. From

the integrated plant safety standpoint, one should not be limited to a specific type of events. In the severe accident prevention section of a safety analysis report, all of the most important threats to the facility and public health and safety (e.g., earthquake and typhoon in the Taiwan area) should be addressed. Please also see Second Round Question 1 on Section 19.1. Will the Lungmen PSAR be updated to include such kind of considerations?

Section 19.3.1.1 – Anticipated Transients Without Scram (ATWS)

1. Table 1 in the response to First Round Question 1 on Section 19.3.1.1 (Track No. 19-025) provides a list of the diverse scram signals by accident initiators. Please provide an explanation for the scram signals associated with feedwater controller failure-maximum demand. Is the low water level signal generated after a high water level trip of the feedwater pumps? What if the high water level trip fails? How is the low vessel level signal generated following initiating event "turbine bypass/control fails closed"? Is the low vessel level signal generated in this event as a result of void collapsing under high reactor pressure condition? Please provide an explanation for the scram signals associated with initiating event "inadvertent rod insert" listed in Table 1? What diverse actuation signals would be timely generated in an ATWS event following a total loss of feedwater? High drywell pressure signal would not be generated if the location of the feedwater line break were outside the primary containment.

2. The response to First Round Question 2 on Section 19.3.1.1 (Track No. 19-025) indicates that large power oscillations can be prevented by a controlled cold water injection following vessel depressurization after a failure of all high pressure injection makeup in an ATWS event. How will the operators control and how difficult it is to control the cold water injection in this scenario? What kind of instructions and training can and will be provided to the operators for controlling cold water injection in this scenario?

Section 19.3.1.2 – Station Blackout (SBO)

1. It is stated in the response to First Round Question 1 on Section 19.3.1.2 (Track No. 19-026) that, with RCIC injection, the number of SRV cycles following an SBO event would be no more than 14. Each accumulator equipped for the SRVs with no ADS function is sized to provide one SRV actuation at normal drywell pressure. The accumulators associated with the ADS SRVs are sized to allow five actuations at normal drywell pressure. Makeup supply to the SRV accumulators is provided by the nitrogen gas bottles. Use of these gas bottles requires manual operator actions at the valve location to open the supply valve. Will the Emergency Operating Procedures (EOP) for the Lungmen NPS or the Emergency Procedure Guidelines (EPG) include provisions to instruct operators to alternate SRV for manual relief operation during the initial RCIC injection following an SBO event? Or, will the EPG/EOP include provision to open the nitrogen supply valve from the beginning of the event sequence?

2. The response to First Round Question 2 on Section 19.3.1.2 (Track No. 19-026) indicates that the Lungmen raceway and enclosure design will include features to minimize ingress of moisture and the impact of moisture that may enter. Will these features be included for all of the equipment required to operate during the response to an SBO event, including the period after AC power restoration? Will design features also be included to compensate for the possible reduction in heat dissipation in certain cases?

3. The response to First Round Question 3 on Section 19.3.1.2 (Track No. 19-026) refers to response to part 1 of Question 19-001, which is not attached. Please provide response to Question 19-001.

4. No response to First Round Question 4 on Section 19.3.1.2 has been received as of May 7, 1998. This question remains to be an open issue.

Section 19.3.1.3 – Fire Protection

1. It is stated in the response to First Round Question 1 on Section 19.3.1.3 (Track No. 19-027) that control and electrical equipment enclosures can provide some protection from water spray. However, different classes of enclosures have been used in nuclear power plants to house control and electrical equipment. Will the design of control and electrical equipment in the Lungmen NPS include only drip-proof and spray-proof enclosures in areas equipped with water suppression system? Similarly, will the design for raceways and their associated junction boxes in these areas be drip-proof and spray-proof?

2. The response to First Round Question 2 on Section 19.3.1.3 (Track No. 19-027) stated that it is highly unlikely that a control or instrumentation circuit could suffer limited damage so that it would short to another conductor, cause an undesired action or indication, and yet remain in service without blowing a fuse or tripping a circuit breaker. Fault detection circuitry, fuses, or circuit breakers can protect equipment from being damaged, but cannot prevent the affected equipment and systems from being unavailable for their design functions needed during plant response to accident initiating events. Fire-induced conductor-to-ground short could cause a fuse blown or a breaker tripped for grounded systems. Conductor-to-conductor short will not necessarily cause the same actions. Hot short has clearly occurred in the Browns Ferry fire for a period of time and caused operator confusion.

   The multiplexing system can essentially prevent spurious control actions, but it cannot prevent the accident prevention/mitigation equipment from losing control signals during a fire event. Will fiber optics multiplexing system be used for all of the control signals? Does the multiplexing system contain dual channels for equipment in each of the three divisions? Are both channels for one safety division independent from the channels for other safety divisions? Are the dual channels separated within the same division? What are the fire-related failure modes and

effects of different multiplexing system channels? Please provide FMEA analysis for the applicable failure modes, if available.

3. The response to First Round Question 3 on Section 19.3.1.3 (Track No. 19-027) indicates that the service water pump house, circulating water pump house, and related tunnels are designed to preclude the possibility of a fire-induced common cause failure causing a simultaneous trip demand or impacting safety equipment on both units. Please explain how will the Lungmen NPS be designed to achieve this. Is the main control room a shared area between the two adjacent units? If it is, please provide information to demonstrate that fire in an area associated with one unit would not affect the control room operation of its adjacent unit.

4. The response to First Round Question 4 on Section 19.3.1.3 (Track No. 19-027) refers to Section AD.3 and Table AD.1-1. The fire compartments defined in Attachment AD of Appendix A are very large areas. For example, each of the three safety divisions in reactor building, control building, and intake structure are broadly lumped together as one fire compartment. Will the definition of fire areas be refined in more detail in the FSAR to allow better identification of the risk significant fire zones? Will the PSAR be updated to include fire initiation frequencies and fire-induced core damage frequencies for risk significant areas in Chapter 19?

Section 19.3.1.4 – Intersystem Loss of Coolant Accident (ISLOCA)

1. The response to First Round Question 1 on Section 19.3.1.4 (Track No. 19-028) refers to Section 3M.3 for the protection of those systems where it is impracticable to upgrade to ultimate rupture strength (URS). Will all of the low pressure sinks listed in Section 3M.3, heat exchangers listed in Section 3M.7, and low pressure piping and components not listed in Section 3MA (by pipeline code) be provided with all three of the alternative protection features listed in this first round question? If not, please provide information to explain the scope of inclusion and the basis for the scope.

2. Section 19.3.1.4: The issue addressed in Round 1 Question 2 (Track No. 19-028, Question 2) relates to the potential for increased coolant temperature in the low pressure piping once the interface boundary such as a check valve or motor operated valve is lost as would be the case for many ISLOCA events. Once this interface is lost, the temperature of the fluid inside the low pressure piping is likely to increase. The strength of the low pressure piping will be sensitive to the pipe or component metal temperatures. It is not clear that the proposed design pressure (i.e., 0.4 times the normal operating RCPB pressure) for the low pressure piping is based on potentially for higher temperatures. Please verify that the low pressure piping design pressure is valid for fluid temperatures that may be higher than those normally experienced by this piping.

3. The response to First Round Question 3 on Section 19.3.1.4 (Track No. 19-028) refers to page A1-14 of Appendix A. This page and the associated section provide no explanation of the frequency of ISLOCA and LOCA outside the containment as

suggested in the response. No other direct response to the question was provided because ISLOCA and LOCA outside the containment were not evaluated in Appendix A.

The response also indicates that the ISLOCA and LOCA outside the containment were not explicitly considered as significant initiators in the PRA of Appendix A because of their low frequencies relative to other LOCAs, and the minor contribution of LOCA to the core damage frequency. ISLOCA and LOCA outside the containment also have very low contribution to the core damage frequency for currently operating plants. The reason that they were analyzed in PRAs/IPEs for the currently operating plants is that the containment is bypassed and the consequence of these events is relatively severe. Based on the frequency of severe release, these events are typically not insignificant.

The frequency of core damage resulting from ISLOCA at Lungmen NPS is expected to be much lower than that for the currently operating BWRs due to the higher design pressure in much of the low pressure piping and components that may interface with the reactor coolant pressure boundary. However, the total frequency of core damage for the Lungmen NPS is also much lower than that for the currently operating BWRs. The relative contribution of ISLOCA to the core damage and severe release frequencies may not necessarily be substantially lower.

Will the ISLOCA (and LOCA outside containment) be quantitatively evaluated in the FSAR and the results included in Chapter 19?

Section 19.4.1 – **Overview of the Containment Design**

1. The response to Round 1 Question 1 (Track 19-029, Question 1) refers to numerous other sections of the PSAR. Section 19.4.1 should address the importance of containment features as they relate to severe accidents. For example, can core debris accumulate in these tunnels during severe accidents and if so, what impact does this accumulation have on suppression pool bypass and overall containment performance. If these access tunnels are not important to severe accidents, the reason that they do not play a significant role in severe accidents it should be so stated. Note that in the response to this question, it is stated that "Section 19.7 will address the containment structural capability and leakage through penetrations during severe accidents." There is currently no information regarding leakage through penetrations in Section 19.7.

2. The Round 1 response to Question 2 of this section (Track 19-029, Question 2) refers to the response to Question 19-011 but no such response has been provided. This question remains to be an open issue.

Section 19.4.3 – **Severe Accident Mitigative Features**

1. Section 19.4 contains a mixture of deterministic analysis and probabilistic analysis. It is not always clear as to how the probabilistic analysis is utilized. Section 19.4.3.3.2 discusses the early containment event tree for HPME and provides an estimate for the probability of early containment failure due to direct containment heating. Section 19.4.3.2.2 discusses decomposition event trees for debris coolability and pedestal resistance to CCI but it does not appear that any probabilistic results are provided. TPC should clearly indicate the purpose of the decomposition event tree discussions in this latter section.

2. Section 19.4.3.1.1: In the discussion of the sensitivity of hydrogen generation to the corium blockage model, it is noted that disabling the blockage and eutectic cutoff models increases the fraction of zirconium oxidized from a base case value of 6.3 % with the models enabled to 15.8 % with the models disabled for the low pressure melt sequences and from 5.1 % to 35.9 % for the high pressure melt sequences. Why is the base case value for high pressure sequences less than the base case value for the low-pressure case? Why is there apparently no dependency on ACIWA operation?

3. Section 19.4.3.1.1: The following table summarizes the results relative to the time to reach rupture disc failure pressure:

	Time to Reach Rupture Disc Pressure (hr)	
Scenario	Channel Blockage (Base Case)	No Blockage
Low Pressure:		
- ACIWA Operating	31.1	30.6
- ACIWA Not Operating	20.2	16.7
High Pressure:		
- ACIWA Operating	25.0	19.7
- ACIWA Not Operating	18.1	7.1

A review of this table indicates the need for greater definition of the accidents being considered. In all cases, the times to reach rupture disc pressure is greater for the cases with ACIWA operating, implying that less steam is being produced and/or the containment atmosphere is cooler. For the base case scenarios, what is the status of containment heat removal? Also, how is the ACIWA flow being delivered to the drywell (i.e., via the vessel or the drywell spray headers)?

4. Section 19.4.3.8.2: In the first paragraph of this section, it is stated that the shield for the EDS will be solid except for the inlet and outlet piping that will go through its roof. In the fourth paragraph of this section, it is stated that both shields have provisions in their roofs to allow water to flow into the sumps when the lower drywell is flooded. These two statements appear to be contradictory. Please clarify this apparent discrepancy.

5. Section 19.4.3.3.2.3: In reviewing Section 19.4.3.3.2.3 and Figure 19.4-18, it is not clear how the probability of containment failure is being calculated. It appears as though the median drywell failure pressure is being compared to the PDFs for the peak drywell pressure for the various initial drywell pressures; however, the results shown on page 19.4-38 do not agree well with this assumption. Please identify the process for obtaining the conditional probabilities for drywell head failure listed on page 19.4-38. Assuming that the reviewer's understanding is correct, then the statement in the original question that potential failures at less than the median drywell head failure pressure is correct. In NUREG/CR-6338, a stress-strength interference approach is used. It does not appear that such an approach was used for the Lungmen containment. In this approach, the probability of containment failure is determined by the area lying under the intersection of the two PDFs. Neglecting the strength distribution below the median failure pressure could result in optimistic conditional probabilities of drywell failure. Have any calculations been performed to evaluate this potential optimism?

   Note that the above question and the following question are followups to Round 1 Question 11 (Track No. 19-033, Question 2). The second part of the original question addressed the impact of COPS on pressure spikes. The question should probably have been worded as follows:

   > Does operation of COPS significantly affect the peak pressures attained in the containment for accident scenarios involving rapid increases in containment pressure?

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-020

PSAR Sections: 19.0

Question Date: March 2, 1998

PSAR Question:

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

Response:

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-021

PSAR Sections: 19.1

Question Date: March 2, 1998

PSAR Question:

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

PSAR Response:

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

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

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

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

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

SECOND ROUND OF QUESTIONS ON THE LUNGMEN PSAR CHAPTER 19 SEVERE ACCIDENT ANALYSIS

Track # 19-021

Section 19.1 Introduction

MPR Question

1. The response to First Round Question 1 on PSAR Section 19.1 (Track No. 19-021) indicates that an SRP addressing severe accident analysis, Chapter 19, specifically developed for Lungmen NPS is attached to the response to the First Round Questions. However, this attachment of the "SRP Chapter 19" cannot be found in the response to the first round questions. The response to Question 1 on PSAR Section 19.1 also refers to the response to Question 19-016 for the basis of PSAR Chapter 19 contents and organization. This "Response to Question 19-016" is also not included in the response to the first round questions. Please provide the "SRP Chapter 19" used in the preparation of Lungmen PSAR and "Response to Question 19-016".

   The USNRC Final Safety Evaluation Report (FSER) for the Standard ABWR was prepared for the certification of a generic design without considering a specific site. The contents and key severe accident issues will not be necessarily the same for the application of a construction permit at a specific site. For instance, regarding the deterministic assessment of severe accident prevention, plant features intended for the response to and/or mitigation of both internal and external hazards should be summarized. The need to address the external hazards is especially important at sites with significant, known natural hazards.

   The content of PSAR should include a preliminary analysis and evaluation of the design and performance of the facility with the objective of assessing the risk to public health and safety (10CFR 50.34) including determination of:

   - The margins of safety during normal operations and transient conditions, and
   - The adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents.

   From the standpoint of public health and safety, the considerations of accidents should not be limited to those initiated from the internal hazards only.

   Even though some of the design information is already provided, in detail in other sections of the PSAR, information relevant to severe accident preventions and mitigation should be summarized both deterministically and probabilistically in an integrated manner in Chapter 19. For each of the safety functions, conformance to the defense-in-depth philosophy as well as redundancy, independence, and diversity requirements should be discussed. Probabilistic measures of the plant's ability to prevent and mitigate

severe accidents (e.g., reliability of safety functions and conditional likelihood of core damage for various accident initiators) should also be provided in Chapter 19.

In addition, differences in design and operations between the Lungmen NPS and Standard ABWR that are relevant to severe accident prevention and mitigation should be summarized. The impact of these differences on Lungmen's ability to prevent and mitigate severe accidents should also be addressed in Chapter 19.

Will the Lungmen PSAR be updated to include Chapter 19 the additional items discussed in the preceding?

Response

1. The SRP and response to Track No. 19-016 are included as attachments to the current response. Also attached is the response to Question 19-015 which discusses the scope of Chapter 19 relative to the remainder of the PSAR. As noted in the latter response, "The whole PSAR may be considered as an account of how the severe accident prevention and mitigation assurance philosophy of page 19.1.1 is applied in the Lungmen NPS design and operation. For this reason, the PSAR Chapter 19 SRP (Reference 19.1-4) confines the scope of Chapter 19 to only those issues discussed in Sections 19.3 through 19.7."

The Lungmen PSAR complies with 10CFR50.34 and the USNRC SRPs applicable to PSAR sections . In particular, the PSAR complies with the "Additional TMI-related requirements" of 10CFR50.34(f), including a plant/site PRA which is documented in Appendix A of the PSAR. Lungmen-site specific hazards are evaluated both deterministically in Chapter 3 of the PSAR and probabilistically in Appendix A. Please see response to Track No. A-009 (attached) for a discussion of how PRA impacted the design of the Standard ABWR and the major modifications made to the design for Lungmen.

Section 19.4 of the PSAR will be updated as described in the response of Track No. 19-016. As indicated in the response to Track No. 19-015, the overview of Section 19.2, pages 19.2-1 through 19.2-4, presents a preliminary summary that aims at placing the contents of Chapter 19 in the context of the overall approach to meet the severe accident protection requirements of page 19.1-1. The above preliminary summary will be updated in the FSAR where it will be supplemented by a table that shows direct relationship between the severe accident protection requirements, the defense-in-depth provisions for meeting these requirements, and where these provisions are covered in the FSAR. Differences in design and operations between the final Lungmen NPS design and Standard ABWR that are relevant to severe accident prevention and mitigation will also be summarized in the FSAR Chapter 19.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-015

PSAR Sections: Ch 19

Question Date: December 26, 1997

PSAR Question:

1. P.19.2-1 listed four key elements and the third one mentioned that further requirements are provided. An example was given in the 2nd paragraph from the last of the same page. However, all the features listed in pages 19.2-2 and 19.2-3 did not identify which ones are used to satisfy the requirements of the 3rd key element. Please explain.

2. The Overview in P.19.2-4 should add a summary description on how the three guarantees and one minimization requirement listed in P.19.1-1 can be met.

PSAR Response:

1. The added selected requirements of the third key element of P.19.2-1 have been derived from consideration of severe accidents that involve reactor vessel melt-through or are accompanied by suppression pool bypass. The following features are presented in pages 19.2-2 through 19.2-4 as examples of design features specifically added to meet these requirements:

   1) Intersystem LOCA prevention provisions (1)(b), page 19.2-2
   2) Large lower drywell floor area to enhance core debris coolability (2)(c), page 19.2-3
   3) ACIWA, for containment cooling and radioactive material removal (2)(d)(ii), page 19.2-4
   4) Lower drywell flooder for debris cooling (2)(d)(iii), page 19.2-4
   5) COPS for controlled pressure relief (2)(e), page 19.2-4.

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

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

(Reference 19.1-4) confines the scope of Chapter 19 to only those issues discussed in Sections 19.3 through 19.7.

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

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-016

PSAR Sections: Ch 19

Question Date: December 26, 1997

PSAR Question:

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

PSAR Response:

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

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

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

- Hydrogen generation and control (Section 19.4.3.1)
- Core debris coolability (Section 19.4.3.2)
- High pressure core melt ejection (Section 19.4.3.3)
- Containment vent design (Section 19.4.3.4)
- Suppression Pool bypass (Section 19.4.3.5)

- Fuel-coolant interaction (Section 19.4.3.6)
- Equipment survivability (Section 19.4.3.7)
- Protection of containment sumps (Section 19.4.3.8).

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

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

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

1) MAAP-code analysis leads to metal-water reaction involving less than 40% of the cladding.
2) Use of the MAAP analysis assumptions indicated above increases the radioactive release if the ac-independent water addition system (ACIWA) is not initiated. However, this increase virtually diminishes if the ACIWA is initiated. Therefore, the ACIWA provides a protection against uncertainty in the fraction of metal reacting.
3) The non-mechanistic hydrogen release corresponding to reaction of 100% of the zirconium cladding increases the containment pressure to 0.618 MPa which is below the 0.72 MPa setpoint of the containment overpressure protection system (COPS) and the 0.77 MPa level C capability of the containment. Therefore, the COPS elevated setpoint and containment structural capacity provide two more levels of safety assurance against the uncertainty in the fraction of metal reacting.

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

Section 19.4.3.2 discusses the core debris coolability potential and its impact on core concrete interaction (CCI). Core-concrete interaction could cause vertical erosion of the containment concrete floor, radial erosion of the pedestal walls, and containment pressurization due to the generated heat and released gases. The section provides an overview of the CCI preventive and mitigative features. These include: 1) a large containment floor area to enhance debris coolability, 2) ACIWA and the lower drywell passive flooder which provide both cooling capability and fission product scrubbing, 3) basaltic concrete in the containment floor to reduce the gas release from CCI, 4) COPS to protect against containment overpressurization, 5) large depth of the basaltic concrete floor to protect the containment liner, and 6) thick pedestal walls to ensure its load bearing capability with CCI erosion. The section also presents analyses of the CCI prevention and mitigation capability. Due to the large number of possible CCI scenarios and uncertainties, both probabilistic and sensitivity analyses are performed. Probabilistic analysis uses decomposition event trees (DET) to establish debris coolability scenarios and their probabilities, and pedestal attack scenarios and their probabilities. The DETs identify the key factors that affect these scenarios, e.g., the

initial amount of corium released to the lower cavity, the level of corium superheat, if and when cooling water is available for debris quenching, upward heat flux, and whether the debris flows to the suppression pool. Sensitivity to various modeling assumptions is investigated using the MAAP-ABWR code. Finally, the section provides evaluations of the pedestal strength and the effect of using different types of concrete for the pedestal walls. These analyses lead to the following main conclusions:

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

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

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

1) The conditional probability of early containment failure due to DCH given core damage is 1×10^{-3}. This low probability presents the combined effects of the ADS,

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

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

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

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

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

Section 19.4.3.5 describes the investigation of suppression pool bypass, e.g., due to containment isolation failure, as a containment failure mode. The section discusses the suppression pool bypass prevention features, develops bypass leakage scenarios, and estimates the corresponding consequences. An extensive screening investigation of the suppression pool bypass pathways (containment isolation lines, ex-containment LOCA, wetwell/drywell interface) shows that the wetwell/drywell vacuum breaker leakage

presents the only potentially risk significant bypass. Probabilistic analysis using DETs is performed to identify leakage scenarios and probabilities. The DETs account for uncertainties in the vacuum breaker leak size and aerosol plugging of leakage paths. The section also presents MAAP-ABWR analysis of the radioactive release to the environment for five accident scenarios involving vacuum breaker leakage. The above analyses lead to the following main conclusions:

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

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

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

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

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

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

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-022

PSAR Sections: 19.2

Question Date: March 2, 1998

PSAR Question:

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

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

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

PSAR Response:

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

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

There are10 TBPVs and 18 SRVs in each of the Lungmen ABWR units. The SRVs are distributed between the four MSLs (4, 5, 5, and 4 SRVs in MSL A, B, C, and D respectively). Eight of the SRVs are assigned to the Automatic Depressurization System (ADS). Each MSL contains 2 ADS valves which are separated within a MSL by a non-ADS valve.

The SRVs provide three pressure control modes:

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

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

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

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

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

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

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

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-023

PSAR Sections: 19.3

Question Date: March 2, 1998

PSAR Question:

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

PSAR Response:

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

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

Section 19.3 Assessment of Severe Accident Prevention

MPR Question

1. Is the swing EDG the only difference between Lungmen NPS and Standard ABWR that is relevant to severe accident prevention? If not, please provide a complete list of the differences in design and operations between the Lungmen NPS and Standard ABWR that are applicable to severe accident prevention. Furthermore, please also provide an evaluation of the effects of these differences on severe accident prevention and mitigation.

Response:

1. Please see attached response to Track No. A-009, item 3.

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: A-009

PSAR Sections: Overview, A

Question Date: January 3, 1998

PSAR Question:

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

Response:

3. The Standard ABWR final design is the product of several years of optimization effort to meet both qualitative risk reduction goals such as design simplification and quantitative risk reduction goals. Specifically, in the early phase of the design, PRA was used to guide design decisions in the following areas:

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

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

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

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

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

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

Robustness of the ABWR design was demonstrated in the Lungmen NPS internal events PRA by importance analysis and sensitivity analysis which did not uncover any particular design weakness. However, the seismic PRA identified components

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-024

PSAR Sections: 19.3.1

Question Date: March 2, 1998

PSAR Question:

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

Response:

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-024

PSAR Sections: 19.3.1

Question Date: March 2, 1998

PSAR Question:

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

   - Frequencies of accident initiators
   - Conditional likelihood of safety function failure given occurrence of an accident initiator. This can be further divided into:
     - Conditional failure likelihood of the scram function given occurrence of an accident initiator.
     - Conditional failure likelihood of the pressure relief function given occurrence of an accident initiator (with success and failure of the scram function separately).
     - Conditional likelihood of a transient-induced LOCA given occurrence of an accident initiator.
     - Conditional failure likelihood of the high pressure injection function given occurrence of an accident initiator and successful scram function (with and without transient-induced LOCA).
     - Conditional failure likelihood of the vessel depressurization function given occurrence of an accident initiator, successful scram function, and failure of the high pressure injection function (with and without transient-induced LOCA).
     - Conditional failure likelihood of the low pressure injection function given occurrence of an accident initiator, successful scram function, failure of high pressure injection, and successful vessel depressurization.
     - Conditional failure likelihood of the decay heat removal function given occurrence of an accident initiator and successful scram function (with vessel water inventory being maintained separately by high pressure injection and low pressure injection).
     - Conditional likelihood of core damage given occurrence of an accident

initiator.

- Conditional likelihood of core damage given a transient-induced LOCA.

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

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

PSAR Response:

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

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

SECOND ROUND OF QUESTIONS ON THE LUNGMEN PSAR CHAPTER 19 SEVERE ACCIDENT ANALYSIS

Track No. 19-024

Section 19.3.1 Severe Accident Preventive Features

MRP Question

1. The response to First Round Question 1 on Section 19.3.1 (Track No. 19-024) indicates that Unresolved Safety Issue (USI) A-45 is resolved for the Lungmen NPS. It is true that A-45 was considered resolved for plants that have completed an Individual Plant Examination (IPE) or a risk assessment and have demonstrated that there are no vulnerabilities related to shutdown decay heat removal (DHR). Contributions to core damage and severe release frequencies due to failure of the DHR function must be acceptably low. Major contributors to core damage or severe release due to DHR function failure should be explicitly identified. In addition, cost-effective corrective actions for these major contributors, if any, should have been evaluated. While the total core damage and severe release frequencies at the Lungmen NPS are less than the USNRC safety goal guidance, it would still be better documentation to provide a separate section using results from the Lungmen PRA to demonstrate that there are no vulnerabilities related to DHR function failure as was done in most of the U.S. IPE programs (i.e., no individual DHR failure contributors dominate the core damage or severe release risk). Will the Lungmen PSAR be updated to include the above information in a separate section?

2. The response to First Round Question 2 on Section 19.3.1 (Track No. 19-024) refer to Appendix A and Attachments AA and AB of Appendix A. However, most of the probabilistic parameters listed in this question (e.g., likelihood of safety function failure) cannot be found in Appendix A and its attachments. This question remains to be an open issue. Will the Lungmen PSAR be updated to present these probabilistic parameters in Chapter 19?

3. The response to First Round Question 3 on Section 19.3.1 (Track No. 19-024) indicates that external events, including earthquake and typhoon are outside the scope of Chapter 19 according to the "Lungmen Chapter 19 SRP". The document referred to in this response, i.e., Lungmen Chapter 19 SRP, is not a "Standard Review Plan" issued by any regulatory agency. This question remains to be an open issue. From the integrated plant safety standpoint, one should not be limited to a specific type of events. In the severe accident prevention section of a safety analysis report, all of the most important threats to the facility and public health and safety (e.g., earthquake and typhoon in the Taiwan area) should

be addressed. Please also see Second Round Question 1 on Section 19.1. Will the Lungmen PSAR be updated to include such kind considerations?

Response

1. As indicated in the response to Track No. A-009, Item 3, the Standard ABWR final design is the product of several years of optimization effort to meet both qualitative and quantitative risk reduction goals. In particular, PRA was used during the early phase of the design to guide the design of several systems including the core cooling systems. As indicated in the response to the first round of this question, the USNRC did not require USI A-45 issue to be addressed in the SSAR.

   The Lungmen design took advantage of the above Standard ABWR optimization effort, with modifications as noted in the response to A-009(3). Please refer to response to Track No. A-009 (3) for discussion of the differences between the Standard ABWR and Lungmen designs and robustness of the Lungmen design under internal and external events. Please refer also to PSAR Chapter 1, Appendix 1C, Section 1C.2.10 for resolution of USI A-31 (RHR Shutdown Requirements) in Lungmen and the RHR features to assure its high reliability.

   The Lungmen PRA is documented in Appendix A of the PSAR. Success criteria for reactor shutdown and heat removal functions are defined in the Appendix Table A4-8 which illustrates the defense-in-depth provisions implemented in Lungmen heat removal capability against various accident initiating events. Figure AA.2-15 contains the fault tree for the DHR and the Executive Summary (Section A1) provides an overview of the Lungmen risk characteristics and risk important systems, structures, components and human errors for internal and external events evaluated in the Lungmen PRA.

   In view of the PRA-based design optimization of the Standard ABWR, and the Lungmen modifications identified in A-009 to meet the Lungmen site-specific challenges which have been based on the PRA of PSAR Appendix A, duplicating the PRA information in a separate section in Chapter 19 is seen as unnecessary and could lead to inconsistencies between Chapter 19 and Appendix A as the PRA is updated.

2. Although the PRA in Appendix A and its attachments do not provide explicit estimates of the specific conditional failure probabilities presented in this question, the PRA identifies the risk-important systems, structures, components and human errors. This is of prime importance for two reasons:

i) The identified risk significant items are used as input to the Lungmen Integrated Reliability Assurance Program to ensure that the assumptions used in the PRA for these critical items will not be violated during operation.

ii) Risk-significant support systems whose failure may fail more than one system are not overlooked.

The Executive Summary of the PSAR Appendix A presents the identified risk critical items for internal and external events. In particular, the risk significance of support systems is clearly demonstrated in Tables A1-5 through A1-7 for risk at 100% power and Tables A1-16 through A1-18 for risk during shutdown.

While the conditional probabilities identified in the Round-1 questions will provide further insight in the Lungmen risk characteristics, they do not adequately portray the risk significant items needed for risk management. Furthermore, a significant effort will be required to obtain these probabilities from the risk model. Consequently, updating the PSAR Appendix A to include these probabilities is not considered cost effective.

3. Response to this question was sent June 10, 1998

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-025

PSAR Sections: 19.3.1.1

Question Date: March 2, 1998

PSAR Question:

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

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

PSAR Response:

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

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

For an ATWS event, a requirement to shutdown the reactor with the reactor not shutdown causes entry into the plant's Emergency Operating Procedures (EOPs). EOPs provide instructions to insert control rods, control reactor pressure and water level, and inject boron if necessary to shutdown the reactor. For an ATWS with HPCF failure following failure of control rod insertion (and Alternate Rod Insertion), as well as all other attempts to insert

enough control rods within the three minute time period, the Lungmen ATWS logic will continue to sense that the reactor power is not low (Startup Range Neutron Monitor sensed power level ≥ 6% power and High Reactor Pressure for three minutes) and will automatically inject boron with SLC initiation.

With a failure of all high pressure makeup, EOPs would require that the reactor be depressurized so that low pressure systems may be used to control reactor water level. EOPs based on the BWR Owners' Group Emergency Procedure Guidelines (EPGs) specifically address the potential for cold water related power excursions for scram failure events by limiting the rate of cold water injection (e.g., LPCF injection) during and following reactor depressurization. The purpose of limiting the rate of cold water injection is to take advantage of the negative reactivity feedback from the increased void fraction in the core region. Assessments of these conditions to support EPG development concluded that a short-term power surge may occur, but it would not develop into a reactor instability (oscillation) even if borated water was not injected by SLC into the core. Large power oscillations may develop at low flow and high power conditions, but with sufficient core void fraction or boron in the reactor there is not sufficient power to produce a large power oscillation due to controlled cold water injection.

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

RESPONSES TO MPR's PSAR QUESTIONS

TABLE 1 PSAR Question 19-025

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

misc. plant occurrences										
MSIV closure (all)	B		B			X				
Loss of Feedwater				X						
Recirculation Pump Trip (one, multiple, all)		X						O		7
Recirculation Flow Control Failure- Increasing Flow	X									8
Recirculation Pump Failure										7
Rod Withdrawal Error at Low Power										9
Rod Withdrawal Error at Power										10
Inadvertent Startup of Idle Recirc. Pump.										16

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

TABLE 1 PSAR Question 19-025 (Continued)

Initiating Events	Scram Sensors For Initiating Events									
	APRM	Core Flow Rapid Coastdown	Reactor Pres. Hi	Reactor Low Level	Drywell Hi Pres.	MSIV Closure	Supp. Pool Hi Temp.	TSV Closure	TCV Fast Closure	Remarks
Control Rod Drop										11
Control Rod Ejection										12
Inadvertent HPCF Startup										13
Feedwater Flow In Core										14
Small Line Break Inside or										15

D 49

Outside Containment										
Steamline Break Outside Containment				B		X				
RWCU Break Outside Containment				X						
Interfacing System LOCA				X						
DBA LOCA (spectrum)				X	B					
RPV Rupture				X	B					

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

RESPONSES TO MPR's PSAR QUESTIONS

Table 1 PSAR Question 19-025 Remarks:

1. Because this event is very slow, the operator action or automatic SCRRI will terminate this event. A loss of 55.6° C feedwater temperature is analyzed to bound this event.
2. Automatic opening of the turbine bypass valves during load rejection or turbine trip will inhibit automatic reactor scram and RPT.
3. A closure of a single MSIV at any given time will not initiate a reactor scram directly. Credit is taken for the operation of reactor high pressure or flux signals to initiate a reactor scram if reactor power is greater than 80% power.
4. PSAR section 15.2.5.3.1. is based on Standard ABWR design, reactor scram is not expected to occur on turbine trip with bypass on condenser low vacuum.
5. Reactor is expected to scram on low reactor level for Lungmen due to loss of Condensate Pumps, Lungmen PSAR Table 15.2-16 analysis event is based on Standard ABWR design.
6. This event describes about the control rod removal error during refueling. The event considers the possibility of inadvertent criticality due to complete withdrawal or removal of the most reactive rod during refueling. The refueling interlock will assure the core shutdown margin is maintained.
7. Reactor scram will be initiated after tripping of all RIPs. Tripping of one or three RIPs, or one RIP seizure event will not cause the reactor to scram, core power will be settled at its steady state after the event.
8. This event assumes fast runout of all RIPs. Reactor power will be settled at new steady state if one RIP experienced a fast runout.
9. The startup range neutron monitor (SRNM) has a period-based trip function that stops continuous rod withdrawal by initiating a rod block. The period-based trip function also initiates a scram if the flux excursion generates a period shorter than 10 seconds.
10. The Automatic Thermal Limit Monitoring (ATLM) operating thermal limit protection function of either MCPR or MAPLHGR protection algorithm stops the rod withdrawal or core flow increase when either operating limit is reached. There is no operating limit violation due to this preventive function.
11. The performance of the FMCRD separation-detection devices and the rod block interlocks virtually preclude the cause of a rod drop accident.
12. The FMCRD brake mechanism prevents the rod ejection event from occurring.
13. This event is a mild transient. The flux level settles out slightly below operating level during this event.
14. These events are presented in decrease in reactor coolant temperature and increase in reactor pressure analysis.
15. This event is bounded by the postulated events. Termination of the analyzed event is dependent on operator action. The action is initiated with the discovery of the unisolable leak. The action consists of the orderly shutdown and depressurization of the reactor vessel.
16. This transient assumes the interlock fails to prevent restart of the RIP. The overcurrent protection logic trips the electrical bus. One or two more RIPs are tripped due to the bus trip. Reactor power settles at the steady state after the RIPs have been tripped.
17. These events are not specifically described in Chapter 15 of PSAR.
18. This event assumes that one of the control valves fast closed at power.

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-025

PSAR Sections: 19.3.1.1

Question Date: March 2, 1998

PSAR Question:

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

Response:

3. Appendix A of the PSAR shows that of the four analyzed external events (seismic, fire, internal flooding, and typhoon) only ATWS initiated by seismic events has a non-negligible contribution to the CDF of the external event. The CDF due to seismic-induced ATWS is 4.5E-7 / reactor year (Total seismic CDF is 3.14E-6 / reactor year). The primary cause of the seismic ATWS is structural deformation that prevents control rod insertion in the core. The ATWS CDF due to the other three external events is negligible. This is to be expected in view of the high degree of redundancy, diversity, and separation of the reactor shutdown systems and the instrumentation and logic channels.

   The PRA in Appendix A is quantified using minimum cutsets that lead to core damage. The annual frequency of ATWS events, whether they lead to core damage or not, is not readily available from these cutsets. Separate calculations will be performed to provide these frequencies in the updated PSAR.

Track No. 19-025

Section 19.3.1.1 Anticipated Transients Without Scram (ATWS)

MPR Questions

1. Table 1 in the response to First Round Question 1 on Section 19.3.1.1 (Track No. 19-025) provides a list of the diverse scram signals by accident initiators.

   a) Please provide an explanation for the scram signals associated with feedwater controller failure-maximum demand. Is the low water level signal generated after a high trip of the feedwater pumps? What if the high water level trip fails?

   b) How is the low water level signal generated following initiating event "turbine bypass/control fails closed"? Is the low vessel level signal generated in this event as a result of void collapsing under high reactor pressure conditions?

   c) Please provide an explanation for the scram signals associated with initiating event "inadvertent rod insert" listed in Table 1?

   d) What diverse actuation signals would be timely generated in an ATWS event following a total loss of feedwater?

   e) High drywell pressure signal would not be generated in the location of the feedwater line break were outside the primary containment.

2. The response to First Round Question 2 on Section 19.3.1.1 (Track No. 19-025) indicates that large power oscillations can be prevented by a controlled cold water injection following vessel depressurization after a failure of all high pressure injection makeup in an ATWS event. How will the operators control and how difficult is it to control the cold water injection in this scenario? What kind of instructions and training can and will be provided to the operators for controlling cold water injection in this scenario?

Response:

1. Responses to the various items of Question 1 are given below

   a) Feedwater controller failure-maximum demand transient initially causes an increase in the RPV inventory, and a high level trip of the main turbine and feedwater pumps are initiated. Signals for scram are initiated by the main turbine stop valves. The main turbine bypass valves open due to the turbine trip, and scram on turbine stop valve position is inhibited. Since the feedwater pumps have been tripped the inventory in the RPV decreases and the plant will be scrammed on low water level.

   If the bypass valves fail to open the scram will be initiated from the turbine stop valve position.

Due to the low probability of a failure of the triplicated feedwater control system coupled with the failure of the reactor water level high signals the failure of high water trip with concurrent feedwater controller failure to maximum demand event is not analyzed. However, if the high water level trip fails then a slow fill of the RPV will occur. The reactor will continue to generate power until eventually the RPV overfills and a large amount of water is carried to the turbine. The turbine will then trip on high vibration. In addition, the high water level alarm (L7) is generated from different transmitters that the high water level 8 turbine trip. So, the operator would be alerted to the problem.

b) The low water level signal is generated following the fast closure of one turbine control valve (turbine bypass/control fails closed) when the voids collapse following the reactor scram. The scram is initiated on high reactor pressure.

c) Table 1 Note 6 describes a case of rod removal during the refueling outage.

At power, for the inadvertent insertion of a single rod the plant will continue to operate normally. For the very unlikely event of an inadvertent gang rod insertion the reactor will reduce power but continue to operate. This event is very unlikely due to the dual redundant architecture of the RCIS.

d) For the ATWS event with a total loss of feedwater the initial actuation signal would be generated by the Reactor Water level low 3 signals. Water level continues to decrease and ARI and FMCRD run-in are initiated on Level 2. Failure of rod insertion will initiate SLCS when the timer is complete and SRNM ATWS permissive signals are present.

e) Yes, a high drywell pressure signal would not be initiated following the a feedwater line break outside containment. The primary scram signal would be reactor water low level. For the case where the feedwater break occurs in the main steam tunnel the leak detection temperature elements would sense the high ambient temperature and initiate automatic isolation of the MSIVs. A scram signal would be generated from closure of the MSIVs.

2. -- (Response to this question is being prepared)

Track No. 19-025
Section 19.3.1.1 Anticipated Transients Without Scram (ATWS)
MPR Questions

1. Table 1 in the response to First Round Question 1 on Section 19.3.1.1 (Track No. 19-025) provides a list of the diverse scram signals by accident initiators.

   a) Please provide an explanation for the scram signals associated with feedwater controller failure-maximum demand. Is the low water level signal generated after a high trip of the feedwater pumps? What if the high water level trip fails?

   b) How is the low water level signal generated following initiating event urbine bypass/control fails closed"? Is the low vessel level signal generated in this event as a result of void collapsing under high reactor pressure conditions?

   c) Please provide an explanation for the scram signals associated with initiating event nadvertent rod insert" listed in Table 1?

   d) What diverse actuation signals would be timely generated in an ATWS event following a total loss of feedwater?

   e) High drywell pressure signal would not be generated in the location of the feedwater line break were outside the primary containment.

2. The response to First Round Question 2 on Section 19.3.1.1 (Track No. 19-025) indicates that large power oscillations can be prevented by a controlled cold water injection following vessel depressurization after a failure of all high pressure injection makeup in an ATWS event. How will the operators control and how difficult is it to control the cold water injection in this scenario? What kind of instructions and training can and will be provided to the operators for controlling cold water injection in this scenario?

Response:

1. Response sent June 10, 1998

2. The initiation and growth of power oscillations in the reactor core is principally dependent upon the subcooling at the core inlet, the greater the subcooling, the more likely that the oscillations will commence and increase in magnitude. When boron is injected into the reactor, the systems used for control of reactor water level must be operated so as to minimize boron dilution and cold water injection, and to promote boron mixing. Premature boron injection could increase the potential for reactivity excursions. If the reactor is not shutdown, reactor water level must be controlled to minimize core inlet subcooling, thereby preventing or mitigating the consequences of any large irregular neutron flux oscillations induced by neutronic/thermal hydraulic instabilities. The following actions address these concerns:

1. Reducing core inlet subcooling by deliberately lowering reactor water level sufficiently below the feedwater spargers (to place the feedwater spargers in the steam space to promote heating of the relative cold feedwater).

2. Reducing reactor power by deliberately lowering reactor water level by terminating and preventing all injections into the reactor, except for boron injection, RCIC, and CRD.

3. Prioritizing the use of injection systems.

4. Controlled injection of cold water while observing reactor nuclear instrumentation response. Nuclear instrumentation includes LPRMs, SRNMs, PRNMs, and Oscillatory Power Range Monitor.

5. Delaying reactor cooldown until reactor shutdown has been assured by either boron injection or control rod insertion.

6. Raising reactor water level to remix the boron concentration in the reactor.

Specifically, as would be required by EOP contingency for reactor level/ power control, prevent automatic initiation of ADS, if any main steam line is not isolated, bypass low reactor water level interlocks to maintain the main condenser as a heat sink. If reactor power is above 5% or cannot be determined and suppression pool temperature is above the Boron Injection Initiation Temperature, and either an SRV is open or opens or drywell pressure is above the high drywell pressure scram setpoint, then:

- Lower reactor water level, irrespective of any reactor power or reactor water level oscillations, by terminating and preventing all injection into the reactor except from boron injection systems and CRD until either:

    1. Reactor power drops below 5%, or
    2. Reactor water level reaches the top of active fuel, or
    3. All SRVs remain closed and drywell pressure remains below the high drywell pressure scram setpoint.

- If reactor water level is above the maximum power control RPV water level (to be determined for Lungmen) and the reactor is not shutdown, then lower reactor water level to below the maximum power control RPV water level by terminating and preventing all injection into the RPV except from boron injection systems and CRD.

- Maintain reactor water level between the minimum steam cooling RPV water level and either:

    1. If RPV water level was deliberately lowered previously, to the level it was lowered ,or

2. If RPV water level was not deliberately lowered with condensate/feedwater, CRD, RCIC, or LPFL.

- If RPV water level cannot be maintained above the minimum steam cooling RPV water level, Emergency RPV Depressurization is required.

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-026

PSAR Sections: 19.3.1.2

Question Date: March 2, 1998

PSAR Question:

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

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

PSAR Response:

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

   As indicated in the response to Question 19-022, the capacity of the nitrogen bottles is large enough to provide 16 times the nitrogen needed to open the 8 ADS SRVs, and after the ADS valves are open, to supply enough nitrogen to substitute for 7-day leakage from the valves accumulators. It is concluded that there is a sufficient nitrogen capacity for RPV pressure control during an SBO.

2. The raceway and enclosure design includes features to minimize the entrance of moisture,

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-026

PSAR Sections: 19.3.1.2

Question Date: March 2, 1998

PSAR Question:

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

Response:

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

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-026

PSAR Sections: 19.3.1.2

Question Date: March 2, 1998

PSAR Question:

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

Response:

4. SBO CDF is 2.55E-6 /reactor year for seismic events, 7.95E-9 /reactor year for fire, 1.5E-10 / reactor year for typhoon, and is negligible for internal flooding.

   As explained in the response to Question 19-025, item 3, the frequency of SBO (with and without core damage) due to external events is not currently available and will be calculated for the revised PSAR.

Track No. 19-026
Section 19.3.1.2 Station Blackout (SBO)
MPR Questions

1. It is stated in the response to First Round Question 1 on Section 19.3.1.2 (Track No. 19-026) that, with RCIC injection, the number of SRV cycles following and SBO event would be no more than 14. Each accumulator equipped for the SRVs with no ADS function is sized to provide one SRV actuation at normal drywell pressure. The accumulators associated with the ADS SRVs are sized to allow five actuations at normal drywell pressure. Makeup supply to the SRV accumulators is provided by the nitrogen gas bottles. Use of these gas bottles requires manual operator actions at the valve location to open the supply valve. Will the Emergency Operating Procedures (EOP) for Lungmen NPS or the Emergency Procedure Guidelines (EPG) include provisions to instruct operators to alternate SRV for manual relief operation during the initial RCIC injection following an SBO event? Or, will the EPG/EOP include provision to open the nitrogen supply valve from the beginning of the event sequence?

2. The response to First Round Question 2 on Section 19.3.1.2 (Track No. 19-026) indicates that the Lungmen raceway and enclosure design will include features to minimize ingress of moisture and the impact of moisture that may enter. Will these features be included for all of the equipment required to operate during the response to an SBO event, including the period after AC power restoration? Will design features also be included to compensate for the possible reduction in heat dissipation in certain cases?

3. The response to First Round Question 3 on Section 19.3.1.2 (Track No. 19-026) refers to response to Part 1 of Question 19-001, which is not attached. Please provide response to Question 19-001.

4. No response to First Round Question 4 on Section 19.3.1.2 has been received as of May 7, 1998. This question remains to be an open issue.

Response:

1. It is expected that the reactor scram has been initiated during the Station Blackout event, and the reactor operator is expected to use the Emergency Operating Procedure (EOP) to control the reactor after one of the EOP entry conditions has been satisfied. The operator is expected to follow the EOP using the available SRVs and/or ADS SRVs in conjunction with RCIC injection to control the reactor pressure. If the operator receives the low pressure alarm in the nitrogen supply to the ADS Valves, the operator is expected to valve in the standby nitrogen supply from the bottled nitrogen by the station operating procedures. If the in-service bottled nitrogen is depleted, a low pressure alarm will be actuated in the control room to alert the operator to change over to the

backup bottled nitrogen. The ADS SRVs can be opened by either Div. 1 or Div. 2 solenoid and its divisional bottled nitrogen supply. The redundant arrangement provides operational flexibility and enhances the ADS SRV availability during the nitrogen supply change over to bottled nitrogen or the change to backup bottles. Thus, even during the change over of bottled nitrogen supply, nitrogen supply is expected to be available to the ADS SRVs.

2. – (Response to this question is being prepared by B&V)

3. Response sent June 10, 1998

4. Response sent June 10, 1998.

~~Please see response to part 3 of Question 19-024~~

Track No. 19-026
Section 19.3.1.2 Station Blackout (SBO)
MPR Questions

1. It is stated in the response to First Round Question 1 on Section 19.3.1.2 (Track No. 19-026) that, with RCIC injection, the number of SRV cycles following and SBO event would be no more than 14. Each accumulator equipped for the SRVs with no ADS function is sized to provide one SRV actuation at normal drywell pressure. The accumulators associated with the ADS SRVs are sized to allow five actuations at normal drywell pressure. Makeup supply to the SRV accumulators is provided by the nitrogen gas bottles. Use of these gas bottles requires manual operator actions at the valve location to open the supply valve. Will the Emergency Operating Procedures (EOP) for Lungmen NPS or the Emergency Procedure Guidelines (EPG) include provisions to instruct operators to alternate SRV for manual relief operation during the initial RCIC injection following an SBO event? Or, will the EPG/EOP include provision to open the nitrogen supply valve from the beginning of the event sequence?

2. The response to First Round Question 2 on Section 19.3.1.2 (Track No. 19-026) indicates that the Lungmen raceway and enclosure design will include features to minimize ingress of moisture and the impact of moisture that may enter. Will these features be included for all of the equipment required to operate during the response to an SBO event, including the period after AC power restoration? Will design features also be included to compensate for the possible reduction in heat dissipation in certain cases?

3. The response to First Round Question 3 on Section 19.3.1.2 (Track No. 19-026) refers to response to Part 1 of Question 19-001, which is not attached. Please provide response to Question 19-001.

4. No response to First Round Question 4 on Section 19.3.1.2 has been received as of May 7, 1998. This question remains to be an open issue.

Response:

1. – (Response to this question is being prepared)

2. – (Response to this question is being prepared)

3. Response to Question 19-001 is attached

4. SBO CDF is 2.5E-6 /reactor year for seismic events, 7.95E-9 /reactor year for fire, 1.5E-10 / reactor year for typhoon, and is negligible for internal flooding. As explained in the response to Question 19-025, item 3, the frequency of SBO (with and without core damage) due to external events is not currently available and will be calculated for the revised PSAR.

~~Please see response to part 3 of Question 19-024~~

D-64

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-001

PSAR Section: 19. 3

Question Date: January 14, 1998

PSAR Question:

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

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

PSAR Response:

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-027

PSAR Sections: 19.3.1.3

Question Date: March 2, 1998

PSAR Question:

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

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

PSAR Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-027

PSAR Sections: 19.3.1.3

Question Date: March 2, 1998

PSAR Question:

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

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

Response:

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

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

Track No. 19-027
Section 19.3.1.3 Fire Protection
MPR Questions

1. It is stated in the response to First Round Question 1 on Section 19.3.1.3 (Track No. 19-027) that control and electrical equipment enclosures can provide some protection from water spray. However, different classes of enclosures have been used in nuclear power plants to house control and electrical equipment. Will the design of control and electrical equipment in the Lungmen NPS include only drip-proof and spray-proof enclosures in areas equipped with water suppression system? Similarly, will the design for raceways and their associated junction boxes in these areas be drip-proof and spray-proof?

2. The response to First Round Question on Section 19.3.1.3 (Track No. 19-027) stated that it is highly unlikely that a control or instrumentation circuit could suffer limited damage so that it would short to another conductor, cause an undesired action or indication, and yet remain in service without blowing a fuse or tripping a circuit breaker. Fault detection circuitry, fuses, or circuit breakers can protect equipment from being damaged, but cannot prevent the affected equipment and systems from being unavailable for their design functions needed during plant response to accident initiating events. Fire-induced conductor-to-ground short could cause a fuse blown or a breaker tripped for grounded systems. Conductor-to-conductor short will not necessarily cause the same actions. Hot short has clearly occurred in the Brown Ferry fire for a period of time and caused operator confusion.

   The multiplexing system can essentially prevent spurious control actions, but it cannot prevent the accident prevention/mitigation equipment from losing control signals during a fire event. Will fiber optics multiplexing system be used for all of the control signals? Does the multiplexing system contain dual channels for equipment in each of the three divisions? Are both channels for one safety division independent from the channels for other safety divisions? Are the dual channels separated within the same division? What are the fire-related failure modes and effects of different multiplexing system channels? Please provide FMEA analysis for the applicable failure modes, if available.

3. The response to First Round Question 3 on Section 19.3.1.3 (Track No. 19-027) indicates that the service water pump house, circulating water pump house, and related tunnels are designed to preclude the possibility of a fire-induced common cause failure causing a simultaneous trip demand or impacting safety equipment on both units. Please explain how will the Lungmen NPS be designed to achieve this. Is the main control room a shared area between the two adjacent units? If it is, please provide information to demonstrate that fire in an area associated with one unit would not affect the control room operation of its adjacent unit.

4. The response to First Round Question 4 on Section 19.3.1.3 (Track No. 19-027) refers to Section AD.3 and Table AD.1-1. The fire compartments defined in Attachment AD of Appendix A are very large areas. For example, each of the three safety divisions in reactor building, control building, and intake structure are broadly lumped together as one fire compartment. Will the definition of fire areas be refined in more detail in the FSAR to allow better identification of the risk significant fire zones? Will the PSAR be updated to include fire initiation frequencies and fire-induced core damage frequencies for risk significant areas in Chapter 19?

Response:

1. – (Response to this question is being prepared by B&V)

2. – (Response to this question is being prepared by B&V)

3. The Reactor Service Water System of each Lungmen unit is a three-division safety system with divisional mechanical and electrical separation. The service water pumps of this system are separated by 3-hour fire walls. This precludes fire-caused total failure of the RBSW of any single unit or of both units. For the pumps of non-safety systems (turbine building service water, and circulating water), fire protection through prevention, detection, and fire confinement and termination before spreading to significant areas are used to prevent fire-caused trip of the two units.

No PSAR changes are planned in response to this comment.

4. As indicated in PSAR Chapter 9, detailed fire hazard analysis is planned for the final Lungmen design and the results will be reported in FSAR Chapter 9, Appendix 9A. This analysis will be used in the FSAR fire PRA.

   Repeating the same information in different parts of the PSAR or FSAR will be kept at a minimum to reduce the chances of possible inconsistencies. Therefore, Chapter 19 will be revised only to include reference to where in the PSAR fire initiation frequencies and fire-induced core damage frequencies for risk significant areas can be found.

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-028

PSAR Sections: 19.3.1.4

Question Date: March 2, 1998

PSAR Question:

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

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

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

PSAR Response:

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

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

This subarticle prescribes the proper temperature to use in the analysis.

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

Track No. 19-028
Section 19.3.1.4 Intersystem Loss of Coolant Accident (ISLOCA)
MPR Questions

1. The response to First Round Question 1 on Section 19.3.1.4 (Track No. 19-028) refers to Section 3M.3 for the protection of those systems where it is impracticable to upgrade to ultimate rupture strength (URS). Will all of the low pressure sinks listed in Section 3M.3, heat exchangers listed in Section 3M.7, and low pressure piping and components not listed in Section 3MA (by pipeline code) be provided with all three of the alternative protection features listed in this first round question? If not, please provide information to explain the scope of inclusion and the basis for the scope.

2. Section 19.3.1.4: The issue address in Round 1 Question 2 (Track No. 19-028, Question 2) relates to the potential for increased coolant temperature in the low pressure piping once the interface boundary such as a check valve or motor operated valve is lost as would be the case for many ISLOCA events. Once this interface is lost, the temperature of the fluid inside the low pressure piping is likely to increase. The strength of the low pressure piping will be sensitive to the pipe or component metal temperatures. It is not clear the proposed design pressure (i.e., 0.4 times the normal operating RCPB pressure) for the low pressure piping is based on potentially for higher temperatures. Please verify that the low pressure piping design pressure is valid for fluid temperatures that may be higher than those normally experienced by this piping.

3. The response to First Round question 3 on Section 19.3.1.4 (Track No. 19-028) refers to page A1-14 of Appendix A. This page and the associated section provide no explanation of the frequency of ISLOCA AND LOCA outside the containment as suggested in the response. No other direct response to the question was provided because ISLOCA and LOCA outside the containment were not evaluated in Appendix A.

   The response also indicates that the ISLOCA and LOCA outside the containment were not explicitly considered as significant initiators in the PRA of Appendix A because of their low frequencies relative to other LOCAs, and the minor contribution of LOCA to the core damage frequency. ISLOCA and LOCA outside the containment also have very low contribution to the core damage frequency for currently operating plants. The reason that they were analyzed in PRAs/IPEs for the currently operating plants is that the containment is bypassed and the consequence of these events is relatively severe. Based on the frequency of severe release, these events are typically not insignificant.

   The frequency of core damage resulting from ISLOCA at Lungmen NPS is expected to be much lower than that for the currently operating BWRs due to the

higher design pressure in much of the low pressure piping and components that may interface with the reactor coolant pressure boundary. However, the total frequency of core damage for the Lungmen NPS is also much lower than that for the currently operating BWRs. the relative contribution of ISLOCA to the core damage and severe release frequencies may not necessarily be substantially lower.

Will the ISLOCA (and LOCA outside containment) be quantitatively evaluated in the FSAR and the results included in Chapter 19?

Response

1. The low pressure sinks listed in Section 3M.3, heat exchangers listed in Section 3M.7 and pipe lines not listed in Section 3MA are not required to apply the three alternative protection features. Section 3MA identifies the RPCB interfacing piping that is designed to the ultimate rupture strength (URS) pressure. The low-pressure pipe lines not listed in 3MA do not interface with RCPB and therefore the three design requirements identified in Section 3M.2 are not applied to them. The low pressure sinks identified in Section 3M.3 do not apply the ISLOCA requirements in Section 3M.2 for RCPB interfacing piping since they do not need to withstand URS as discussed in Section 3M.3. The heat exchangers listed in Section 3M.7 do not apply the 3 requirements in Section 3M.2 for RCPB criteria since they are designed for URS or isolated from RCPB as discussed in Section 3M.7.

2. The low-pressure URS design pressure is valid for fluid temperatures higher than the normal operating temperatures. The USNRC evaluation of the ultimate rupture strength (URS) of low pressure pipe assumed that the low-pressure piping will approach the RCPB temperatures. (Reference: NUREG - 1503, ABWR FSER Section 3.9.3.1.1)

3. Response sent June 10, 1998

Track No. 19-028

Section 19.3.1.4 Intersystem Loss of Coolant Accident (ISLOCA)

MPR Questions

1. The response to First Round Question 1 on Section 19.3.1.4 (Track No. 19-028) refers to Section 3M.3 for the protection of those systems where it is impracticable to upgrade to ultimate rupture strength (URS). Will all of the low pressure sinks listed in Section 3M.3, heat exchangers listed in Section 3M.7, and low pressure piping and components not listed in Section 3MA (by pipeline code) be provided with all three of the alternative protection features listed in this first round question? If not, please provide information to explain the scope of inclusion and the basis for the scope.

2. Section 19.3.1.4: The issue address in Round 1 Question 2 (Track No. 19-028, Question 2) relates to the potential for increased coolant temperature in the low pressure piping once the interface boundary such as a check valve or motor operated valve is lost as would be the case for many ISLOCA events. Once this interface is lost, the temperature of the fluid inside the low pressure piping is likely to increase. The strength of the low pressure piping will be sensitive to the pipe or component metal temperatures. It is not clear the proposed design pressure (i.e., 0.4 times the normal operating RCPB pressure) for the low pressure piping is based on potentially for higher temperatures. Please verify that the low pressure piping design pressure is valid for fluid temperatures that may be higher than those normally experienced by this piping.

3. The response to First Round question 3 on Section 19.3.1.4 (Track No. 19-028) refers to page A1-14 of Appendix A. This page and the associated section provide no explanation of the frequency of ISLOCA AND LOCA outside the containment as suggested in the response. No other direct response to the question was provided because ISLOCA and LOCA outside the containment were not evaluated in Appendix A.

   The response also indicates that the ISLOCA and LOCA outside the containment were not explicitly considered as significant initiators in the PRA of Appendix A because of their low frequencies relative to other LOCAs, and the minor contribution of LOCA to the core damage frequency. ISLOCA and LOCA outside the containment also have very low contribution to the core damage frequency for currently operating plants. The reason that they were analyzed in PRAs/IPEs for the currently operating plants is that the containment is bypassed and the consequence of these events is relatively severe. Based on the frequency of severe release, these events are typically not insignificant.

   The frequency of core damage resulting from ISLOCA at Lungmen NPS is expected to be much lower than that for the currently operating BWRs due to the

higher design pressure in much of the low pressure piping and components that may interface with the reactor coolant pressure boundary. However, the total frequency of core damage for the Lungmen NPS is also much lower than that for the currently operating BWRs. the relative contribution of ISLOCA to the core damage and severe release frequencies may not necessarily be substantially lower.

Will the ISLOCA (and LOCA outside containment) be quantitatively evaluated in the FSAR and the results included in Chapter 19?

Response

1. – (Response to this question is being prepared)

2. – (Response to this question is being prepared)

3. The ISLOCA (and LOCA outside containment) will be quantitatively evaluated in the FSAR Appendix A. The results will be referenced as appropriate in the FSAR Chapter 19.

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-029

PSAR Sections: 19.4.1

Question Date: March 2, 1998

PSAR Question:

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

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

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

PSAR Response:

1. The purpose of Section 19.4.1 is to provide an overview of the containment design. Access tunnels and their airlocks and closures are part of part of the primary containment boundary, and their integrity is maintained in accordance with PSAR Section 16.3.6. Detailed analysis and discussion of severe accidents and mitigative features including the containment are included in Sections 19.3 through 19.7. In particular, Section 19.7 will address the containment structural capability and leakage through penetrations during severe accidents. Please see PSAR Chapters 3 and 6 for detailed design description, structural design, and design basis accident analysis.

2. Please see response to Question 19-011

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

Track No. 19-029
Section 19.4.1 Overview of the Containment Design
MPR Questions

1. The response to Round 1 Question 1 (Track 19-029, Question 1) refers to numerous other sections of the PSAR. Section 19.4.1 should address the importance of containment features as they relate to severe accidents. For example, can core debris accumulate in the access tunnels during severe accidents and if so, what impact does this accumulation have on suppression pool bypass and overall containment performance. If these access tunnels are not important to severe accidents, the reason that they do not play a significant role in severe accidents it should be so stated. Note that in the response to this question, it is stated that ection 19.7 will address the containment structural capability and leakage through penetrations during severe accidents." There is currently no information regarding leakage through penetrations in Section 19.7.

2. The Round 1 response to Question 2 of this section (Track 19-029, Question 2) refers to the response to question 19-011 but no such response has been provided. This question remains to be an open issue.

Response:

1. Leakage from the access tunnels penetrations, hatches, or airlocks present the only issue relevant to severe accident analysis. Debris accumulation is not an issue since the elevation of the tunnels is much higher than the largest possible debris depth (30 cm). As indicated in Section 19.7, some of the subsections in that section will be provided in the FSAR.

   No PSAR changes are planned in response to this comment.

2. Response to Question 19-011 is attached

RESPONSES TO ROC-AEC's PSAR QUESTIONS

Track Number: 19-011

PSAR Sections: Ch 19.4.3.1

Question Date: November 24, 1997

PSAR Question:

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

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

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

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

PSAR Response:

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

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

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

The core damage frequency (CDF) during shutdown is less than 1% of the total CDF (Table A1-4 of the PSAR Appendix A). As indicated in Page A7-17 of the PSAR Appendix A, 97% of the shutdown CDF is due to events occurring during modes 4 (cold shutdown) and 5 (refueling) where the PRA did not take credit for the containment integrity by assuming that the drywell head is removed. Thus, hydrogen burning during these modes does not impact the risk. Consequently, hydrogen burning will impact the shutdown risk only for accident sequences occurring during

mode 3 (hot shutdown) which contribute 3% of the shutdown CDF (or less than 0.03% of the total CDF) and only if no inerting of the containment is initiated.

The containment is not inerted during startup (mode 2; from shutdown to 15% power) following refueling and unplanned shutdowns that require containment entry. The containment may continue to be de-inerted for up to 24 hours after power is increased above 15% (mode 1). In preparation for shutdown, the containment may also be de-inerted for up to 24 hours prior to decreasing power below 15%. The total time in modes 1 and 2 during which the containment is de-inerted is about 3 days or less. Since refueling is done every 18 months, the average refueling mode 1 and mode 2 time during which the containment is de-inerted is no more than 2 days/year. Unplanned shutdowns are projected for Lungmen to be less than 1 shutdown per year. Assuming that containment entry will be needed in 50% of unplanned shutdowns, the average containment de-inerting time due to unplanned outages is no more than 1.5 days/year (0.5 de-inerted shutdowns/year x 3 days/shutdown). Thus the total (refueling and unplanned outages) mode 1 and mode 2 time during which the containment is not inerted is conservatively estimated as 3.5 days/year or less than 1% of the time of plant operation. Should hydrogen release become a real threat or occur during this time, the containment can be re-inerted within 4 hours to prevent hydrogen combustion. The risk of hydrogen burning during shutdown or startup is further reduced by the use of the hydrogen recombiners.

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

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

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

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

No PSAR revision is proposed in response to this question.

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-029

PSAR Sections: 19.4.1

Question Date: March 2, 1998

PSAR Question:

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

Response:

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

however, the results shown on page 19.4-38 do not agree well with this assumption. Please identify the process for obtaining the conditional probabilities for drywell head failure listed on page 19.4-38. Assuming that the reviewer understanding is correct, then the statement in the original question that potential failures at less than the median drywell head failure pressure is correct. In NUREG/CR-6338, a stress-strength interference approach is used. It does not appear that such an approach was used for the Lungmen containment. In this approach, the probability of containment failure is determined by the area lying under the intersection of the two PDFs. Neglecting the strength distribution below the median failure pressure could result in optimistic conditional probabilities of drywell failure. Have any calculations been performed to evaluate this potential optimism?

Note that the above question and the following question are followups to Round 1 Question 11 (Track No. 19-033, Question 2). The second part of the original question addressed the impact of COPS on pressure spikes. The question should probably have been worded as follows:

Does operation of COPS significantly affect the peak pressures attained in the containment for accident scenarios involving rapid increases in containment pressure?

Response:

1. A summary of Section 19.4 that discusses its organization and contents is provided in response to Track No. 19-016. Please see the above response for a discussion of the analysis in Sections 19.4.3.3.2 and19.4.3.2.2, and for discussion of the issues of HPME, CCI, and use of the decomposition event trees.

2. MAAP3.0B-ABWR results of the high pressure and low pressure base cases suggest that the difference in the generated hydrogen fraction is due to two factors: i) Time to core uncovery, which is longer for the high pressure base case, and ii) Metal/water reaction rate which is slower for the high pressure case than the low pressure case.

The lack of hydrogen generation dependence on the ACIWA operation is due to the definition of the studied accidents where the fire spray is not initiated in time to prevent core damage and RPV failure (Fire spray assumed to be initiated four hours after the accident initiation).

3. The following tables present key fission products release parameters for the four cases discussed in Section 19.4.3.1.1. ~~In the following tables~~ Cases 1 and 2 in Table 1 refer to low pressure core melt scenarios with debris cooling in the containment by ACIWA spray and LDF_1 respectively., ~~and~~ Cases 3 and 4 (Table

2) are high pressure core melt scenarios with debris cooling in the containment by a combined LDF/ACIWA spray and LDF, ~~and a combined LDF/ACIWA spray~~ respectively. Except for Case 4,~~3~~ where absence of containment spray leads to leakage through upper drywell penetrations before COPS rupture disk opens, ~~which assumes suppression pool bypass, the~~ radioactive release in the other cases is through COPS after scrubbing in the suppression pool and occurs when the wetwell airspace pressure increases to the rupture disk set point-.

Detailed definition of the base case accident scenarios and the MAAP3.0B-ABWR analysis of the base cases are described in PSAR Sections AJ.2.1 and AJ.2.2. Please see these sections for more detail. Please see the above sections for the status of the containment heat removal. The ACIWA delivers drywell spray for the accident scenarios presented in Tables 1 and 2

4. The fourth paragraph of Section 19.3.8.2 is incorrect. The provisions in the drain sumps described in the paragraph are on the side (from behind the pedestal wall) and not on the roof. The paragraph will be corrected in PSAR amendment.

5. As stated in the first round response, the conditional probabilities of the drywell failure have been estimated using a stress-strength approach similar to that used in NUREG/CR-6338. Please note that the truncation indicated in Figure 19.4-18 is for pressures lower than the level C pressure of 0.77 MPa and not the median failure pressure of 1.03 MPa. No calculations have been made to evaluate the impact of this truncation but the effect on the failure probability is expected to be negligible.

   Depending on the rapid pressure increase rate, COPS may relieve the containment pressure before containment failure. In the high pressure melt ejection scenarios of Section 19.4.3.3, no credit was taken for the pressure relief by COPS during the period of rapid pressure increase.

Table 1 - Low Pressure Melt Cases

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

Table 2 - High Pressure Melt Cases

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

* Case assumes no containment spray, which leads to leakage through the drywell penetrations.
** Start time for Release through drywell penetrations.

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-031

PSAR Sections: 19.4.3.1

Question Date: March 2, 1998

PSAR Question:

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

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

Response:

1. The analysis in Chapter 19 used the ABWR version of MAAP3.0B. It is true that the MAAP3.0B channel blockage control parameter has been set to the value that disables the blockage and eutectic cutoff models. The following tables present key fission products release parameters for the four cases discussed in Section 19.4.3.1.1. Cases 1 and 2 in Table 1 refer to low pressure core melt scenarios with debris cooling in the containment by ACIWA spray and LDF, respectively. Cases 3 and 4 (Table 2) are high pressure core melt scenarios with debris cooling in the containment by a combined LDF/ACIWA spray and LDF, respectively. Except for Case 4, where absence of containment spray leads to leakage through upper drywell penetrations before COPS rupture disk opens, radioactive release in the other cases is through COPS after scrubbing in the suppression pool and occurs when the wetwell airspace pressure increases to the rupture disk set point.

The MAAP3.0B-ABWR results for the four cases of Tables 1 and 2 below indicate that the assumption of no blockage increases the source term. As seen in these tables, disabling the blockage and eutectic cutoff in the MAAP analysis results in an earlier RPV failure (compared to the blockage case or Base Case) which results in an earlier containment challenge. The earlier containment challenge leads in turn to an earlier release from the containment through COPS (Cases 1 through 3) or upper drywell penetrations (Case 4). The earlier containment release provides less time for radioactivity decay and aerosol settling within the containment, and consequently a higher radioactivity level of released fission gases, and higher release fractions of radioactive aerosols . This is illustrated by the CsI

D-8C

RESPONSX

release fractions with and without blockage shown in the following tables for Cases 1 and 4. The MAAP results of Cases 2 and 3 also show increase in the release fractions when the blockage and eutectic cutoff is disabled in the MAAP analysis. However, the increase in these latter cases is not large enough to increase the CsI release fractions above 1.E-7.

2. For consistency with the practice utilized during the US ABWR Certification Project, an A, G, or D will be applied to the pressure units to indicate absolute, gauge, or differential pressure.

In the PSAR Chapter 19, all MAAP code output pressures and unqualified pressures are in absolute units.

RESPONSES TO MPR's PSAR QUESTIONS

Table 1 - Low Pressure Melt Cases

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

Table 2 - High Pressure Melt Cases

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-032

PSAR Sections: 19.4.3.2

Question Date: March 2, 1998

PSAR Question:

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

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

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

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

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

Response:

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

containment liner. In effect, the faster ablation rate is compensated for by the basaltic concrete depth.

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

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

4. The impact of non-uniform distribution of the debris has been excluded from discussion because it was judged to have second order effect. A discussion will be added to Section 19.4.3.2.2.1 to explain the basis for this conclusion. The discussion will identify potential causes and configurations of non-uniform debris distributions, and the probability and effect of such distributions. The above discussion will be included in the revised PSAR.

5. The basis for Case 3 probabilities may be explained as follows. As stated in page 19.4-18, the range of upward heat flux for this case is from 200 to 400 kW/m^2 It is convenient to divide this range into two parts:

   1) From 300 - 400 kW/m^2. This range is similar to that of Case 2. As indicated in page 19.4-18, the probability of "No CCI" for Case 2 is 1.0.
   2) From 200 - 300 kW/m^2. This range falls between Case 2 and Case 4. Therefore, the probability of "No CCI" for this range is between 0.0 and 1.0. A value of 0.5 for the "No CCI" probability has been chosen as an unbiased value. .

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-032

PSAR Sections: 19.4.3.2

Question Date: March 2, 1998

PSAR Question:

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

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

PSAR Response:

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-033

PSAR Sections: 19.4.3.3

Question Date: March 2, 1998

PSAR Question:

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

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

PSAR Response:

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

2. The conditional probabilities of drywell failure in Section 19.4.3.3 are the same as those developed in the Standard ABWR SSAR. The SSAR containment failure conditional probabilities have been derived using a stress-strength approach. Please see Section AJ.18 of Appendix A for the containment strength distribution. As for the second part of the question, COPS will relieve the drywell pressure if the wetwell airspace pressure reaches the COPS rupture disk set point.

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-034

PSAR Sections: 19.4.3.4

Question Date: March 2, 1998

PSAR Question:

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

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

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

PSAR Response:

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-035

PSAR Sections: 19.4.3.6

Question Date: March 2, 1998

PSAR Question:

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

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

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

PSAR Response:

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-036

PSAR Sections: 19.4.3.7

Question Date: March 2, 1998

PSAR Question:

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

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

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

PSAR Response:

1. The containment pressure of 0.62 MPa of Section 19.4.3.1.2 is based on non-mechanistic, conservative, design basis calculations which assume a design basis LOCA combined with 100% metal water reaction at the same time. The drywell pressure of Figure 19.4-25a was obtained by MAAP code analysis. Although the analysis included conservative modeling assumptions to produce 100% metal water reaction, the results are more realistic in that they considered a TMI-type (isolation) accident with multiple operator actions to turn ECCS on and off to obtain the 100% metal water reaction. As stated in the second paragraph of Section 19.4.3.7.1.1.1, page 19.4-74, a multiplier that effectively increased the rate of metal water reaction was used in the MAAP model. This led to the rapid increase of the containment pressure from ~0.1 to ~0.4 MPa which corresponds to the oxidation of 100% of the Zr cladding. This explains why the pressure remains almost constant (notice that the unit of the time axis is in hours).

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

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

RESPONSES TO MPR's PSAR QUESTIONS

Track Number: 19-037

PSAR Sections: 19.4.3.8

Question Date: March 6, 1998

PSAR Question:

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

PSAR Response:

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

RESPONSES TO MPR's PSAR QUESTIONS

RECOMMENDATIONS/TYPOS

PSAR Sections: 19.4.3.8

Question Date: March 6, 1998

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

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

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

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

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

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

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

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

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

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

RESPONSES TO MPR's PSAR QUESTIONS

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

PSAR Response:

1. All the recommendations will be included in the revised PSAR except as follows:

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

PART III

ADDENDUM 1 TO MPR SER ON PARTIAL REVIEW OF LUNGMEN PSAR

Summary of Open Items

LUNGMEN PSAR REVIEW - SECTIONS 3.7 AND 3.8 OPEN ITEM LIST

Track No.	PSAR Section	Round No.	Question No.	Status
03-020	3.7.1	1	1	Closed
	3.7.1	1	2	Open
	3.7.1	1	3	Closed
	3.7.1	1	4	Closed
	3.7.1	1	5	Closed
03-021	3.7.2	1	1	Closed
	3.7.2	1	2	Open
	3.7.2	1	3	Closed
	3.7.2	1	4	Closed
	3.7.2	1	5	Closed
	3.7.2	1	6	Closed
	3.7.2	1	7	Closed
	3.7.2	1	8	Closed
	3.7.2	1	9	Closed
	3.7.2	1	10	Closed
	3.7.2	1	11	Closed
03-022	3.7.3	1	1	Closed
		1	2	Open
		1	3	Closed
		1	4	Closed
		1	5	Closed
		1	6	Closed
03-023	3.7.4	1	1	Closed
03-024	3.8.1	1	1	Closed
		1	2	Closed
		1	3	Closed
03-025	3.8.1	1	1	Closed
		1	2	Closed
		1	3	Closed
		1	4	Closed

Track No.	PSAR Section	Round No.	Question No.	Status
		1	5	Closed
03-026	3.8.2	1	1	Closed
		1	2	Closed
03-027	3.8.3	1	1	Closed
		1	2	Closed
		1	3	Closed
		1	4	Closed
		1	5	Closed
		1	6(1)	Closed
		1	6(2)	Closed
		1	7	Closed
		1	8	Closed
		1	9	Closed
03-028	3.8.4	1	1	Open
		1	2	Open
		1	3	Closed
		1	4	Closed
		1	5	Closed
		1	6	Closed
		1	7	Closed
		1	8	Closed
		1	9	Closed
		1	10	Closed
		1	11	Closed
		1	12	Closed
		1	13	Closed
		1	14	Closed
		1	15	Closed
		1	16	Closed
		1	17	Closed

LUNGMEN PSAR REVIEW - SECTIONS 3.7 AND 3.8 OPEN ITEM LIST (CONTINUED)

Track No.	PSAR Section	Round No.	Question No.	Status
		1	18	Closed
03-029	3.8.5	1	1	Open
03-046	3.7.1	2	1	Closed
		2	2	Open
		2	3	Closed
03-047	3.7.2	2	1	Closed
		2	2	Open
		2	3	Open
		2	4	Closed
		2	5	Closed
		2	6	Closed
03-048	3.7.3	2	1	Closed
		2	2	Closed
		2	3	Closed
03-049	3.8.1	2	1(a)	Closed
		2	1(b)	Closed
03-050	3.8.4	2	1(a) (b) (c)	Open
		2	2	Closed
03-051	3.8.5	2	1	Open
N/A	3.8.4.2.7 3.8.4.3.6 3.8.4.4.5 3.8.4.5.7	N/A	N/A	Open
N/A	3.8.4	N/A	N/A	Open
N/A	3.8.4	N/A	N/A	Open

Evaluation of Open Items in Lungmen PSAR Sections 3.7 and 3.8 Review

Track No. 03-020, Question No. 2 and Track No. 03-046, Question No. 2

PSAR Section: 3.7.1

The damping ratios used in the analysis of the Lungmen NPS seismic Category I structures and components comply with the SSE damping ratios specified in R.G. 1.61, "Damping Values for Seismic Design of Nuclear Power Plants," and R.G. 1.84, "Design and Fabrication Code Case Acceptability, ASME Section III Division I." The PSAR does not identify how structure stress levels and damping values are to be correlated as specified by the SRP. The use of maximum damping values specified by R.G. 1.61 is acceptable for seismic analysis to obtain seismic loads for structure design since structure stresses cannot exceed permissible limits without these damping values being achieved. However, compatibility between structure stress levels and structure damping values for seismic response analysis to obtain in-structure response spectra for equipment design must be demonstrated. A procedure to provide this compatibility is presented in Section 3.1.2.2 of ASCE 4-86, "Seismic Analysis of Safety-Related Nuclear Structures."

Track No. 03-021, Question No. 2 and Track No. 03-047, Question No. 2

PSAR Section: 3.7.2

Methods for analytical modeling of the seismic Category I structures, RPV and internals, and radwaste building are presented in the PSAR. In general, three-dimensional models are used for seismic analysis of the seismic Category I structures. However, two-dimensional models may be used if coupling between the two horizontal directions is negligible. Three-dimensional models are considered by the staff to be preferable. In response to the staff's questions on the use of two-dimensional models, TPC committed to establishing the equivalence between the two- and three-dimensional models by

comparing only their calculated frequencies and modal participation factors. The staff considers it appropriate to compare mode shapes as well.

Track No. 03-047, Question No. 3

PSAR Section: 3.7.2

The Lungmen PSAR does not describe seismic analysis of the turbine building. TPC provided additional information concerning the main steam lines and turbine building in response to the staff's questions. The main steam lines are to be routed from the reactor building through the main steam tunnel above the control building and into the turbine building. The MS-BOP main steam line piping, bypass line, and condenser are required to remain functional during and after an OBE and a SSE. The MS-BOP main steam lines, turbine bypass valve lines including the turbine bypass valve steam chests and all branch lines 65 mm pipe size and larger, up to and including the first valve that is either normally closed or capable of automatic closure during all modes of operation, are to be dynamically analyzed to demonstrate their structural integrity under OBE and SSE loading conditions. The turbine building is a seismic Category IIA building. Such structures are designed according to ROC Building Code "Medium Seismicity Zone" criteria and are analytically checked to determine that they will not collapse when subjected to an SSE and extreme environmental loads. The two horizontal and one vertical SSE ground motions are considered to act simultaneously when checking the structural integrity of the turbine building.

Dynamic analysis of the turbine building is necessary to determine SSE seismic loads for verification that structural integrity is retained and to obtain seismic input for dynamic analysis of the main steam lines and other associated components. Acceptability of the turbine building seismic analysis cannot be determined since the PSAR does not describe this analysis. Also, the staff cannot determine if the SSE structural integrity check of the turbine building is acceptable since detail on criteria for this check, such as analysis methods, load combinations, and acceptance criteria, have not been provided.

Track No. 03-022, Question No. 2

PSAR Section: 3.7.3

When the time history analysis method is used the input time history is broadened by plus and minus 15% of the integration time step. Force-time histories for non-seismic dynamic loads such as safety relief valve blowdown are not broadened on the basis of conservatism observed between dynamic tests and analysis.

The staff requested justification for not broadening the force-time history for non-seismic loads such as SRV blowdown. TPC responded that tests have been performed which confirm the conservatism of the analytical results. Therefore, for the SRV blowdown loads, the calculated force-time histories are not broadened.

The staff agrees that, if test data confirms the conservatism of the calculated force-time histories, that adding further conservatism by broadening is not warranted. However, the response to the question is general and specific test reports and numerical ratios of test vs. analysis results have not been identified. The PSAR should identify the specific test data source and specify the degree of conservatism.

Track No. 03-028, Question No. 1 and Track No. 03-050, Question No. 1 (a) (b) (c)

PSAR Section: 3.8.4

It was noted in the PSAR Section 3.8.4 review that there are neither descriptions of Lungmen seismic Category I storage tanks (e.g., condensate water storage tank, diesel fuel oil storage tanks, etc.) nor design commitments provided to these tanks. Review questions were raised in both rounds of the PSAR review. TPC has not adequately addressed all the questions asked regarding these tanks. All the seismic Category I yard storage tanks need to be identified in Section 3.8.4 with detailed descriptions of the tanks provided in order to meet the requirements of R.G. 1.70. There should be subsections in Section 3.8.4 dedicated to design commitments for these tanks such as loads and load combinations, design and analysis procedures, structural acceptance criteria, and constructions, etc.

TPC did not respond to Question 1(c) on whether there is any buried (underground) seismic Category I tanks. If such tanks are considered in the design, design commitments should be included in the PSAR.

Track N. 03-028, Question No. 2

PSAR Section: 3.8.4

The last paragraph of Section 3.8.4 states that Lungmen Units 1 and 2 do not contain seismic Category I pipeline buried in soil. The reviewer asked TPC if seismic Category I service water pipelines are enclosed in an underground tunnel. It is the reviewer's experience that service water lines are usually either buried or enclosed in concrete tunnels. TPC responded by stating that safety related seismic Category I piping systems and their supports are designed in accordance with ASME III. This response does address the reviewer's question. Should the Lungmen service water lines be enclosed in reinforced concrete tunnels, design commitments for both the pipelines and the tunnels should be provided in Section 3.8.4.

Track No. 03-029, Question No. 1 and Track No. 03-051, Question No. 1

PSAR Section: 3.8.5

TPC in the response to round 2 review question stated that the Lungmen foundation construction is a multi-layered approach. On top of the intact rock a layer of 100 mm thick mudmat will be placed. The mudmat will have drain pipes installed and its surface intentionally roughened. A waterproofing membrane will be laid on top of this mudmat. On top of the membrane there will be another layer of 100 mm thick mudmat. The foundation basemat will be placed on top of this mudmat.

Frictional coefficients between mudmat and bedrock and between waterproofing membrane and concrete were presented by TPC. A value of 0.75 was specified for the interface between the mudmat and the waterproofing membrane. Kuosheng FSAR along with "various text books" were referenced for selecting this value. It is reviewer's understanding that Kuosheng NPS has a dewatering system. The reviewer is not aware of a similar dewatering system for Lungmen, since the review scope is limited to Sections 3.7 and 3.8 of the PSAR. If such a system is not present, the effect of

groundwater on the friction coefficient between the mudmat and the waterproofing membrane needs to be addressed by TPC.

Track No. N/A, Question No. N/A

PSAR Sections: 3.8.4.2.7, 3.8.4.3.6, 3.8.4.4.5, and 3.8.4.5.7

For the safety service water pump house, Lungmen PSAR provided a paragraph of description of the structure in Subsection 3.8.4.1.7. Other PSAR information concerning applicable codes, standards and specifications, loads and load combinations, design and analysis procedures, and structural acceptance criteria for the pump house are to be provided in the FSAR as stated in PSAR Subsections 3.8.4.2.7, 3.8.4.3.6, 3.8.4.4.5, and 3.8.4.5.7, respectively. It is likely that the design considerations of the safety service water pump house will be similar to that of other seismic Category I structure. As a minimum, design commitments should be stated in the PSAR.

Track No. N/A, Question No. N/A

PSAR Section: 3.8.4

A subsection which provides information on the materials used in the construction of the reactor building, control building, auxiliary fuel building, and radwaste building needs to be provided in the PSAR. The subsection should include information such as major materials of construction (e.g., reinforced concrete, structural steel and anchors), quality control, special construction techniques, and quality assurance.

Track No. N/A, Question No. N/A

PSAR Section: 3.8.4

The turbine building is a seismic Category IIA structure. Per definition in Table 3.2-1d, seismic Category IIA are SSC whose collapse could result in loss of required function of seismic Category I structures, components, or systems required for safe shutdown. Thus, the turbine building must be capable of withstanding the Lungmen Safe Shutdown Earthquake (SSE) so as not to impair safety function of a portion of the main steam line, the condenser, or the adjacent seismic Category I structures. There are currently no

discussions in Section 3.8.4 concerning design commitments for the turbine building. Subsections on codes and standards, loads and load combinations, structural acceptance criteria, design and analysis procedures of the turbine building should be provided.

LUNGMEN PSAR REVIEW – CHAPTER 19 OPEN ITEM LIST

Track No.	PSAR Section	Round No.	Question No.	Status
19-020	19.0	1	1	Closed
19-021	19.1	1	1	Closed
		2	1	Open
19-022	19.2	1	1	Closed
			2	Closed
			3	Closed
19-023	19.3	1	1	Closed
?	19.3	2	1	Open
19-024	19.3.1	1	1	Open
		2	1	Open
		1	2	Open
		2	2	Open
		1	3	Open
		2	3	Open
19-025	19.3.1.1	1	1	Closed
		2	1	Closed
		1	2	Closed
		2	2	Open
		1	3	Closed
19-026	19.3.1.2	1	1	Closed
		2	1	Closed
		1	2	Closed
		2	2	Open
		1	3	Closed
		2	3	Closed
		1	4	Closed
		2	4	Closed

Track No.	PSAR Section	Round No.	Question No.	Status
19-027	19.3.1.3	1	1	Closed
		2	1	Open
		1	2	Closed
		2	2	Open
		1	3	Closed
		2	3	Open
		1	4	Closed
		2	4	Closed
19-028	19.3.1.4	1	1	Open
		2	1	Open
		1	2	Closed
		2	2	Closed
		1	3	Closed
		2	3	Closed
19-029	19.4.1	1	1	Open
		2	1	Open
		1	2	Closed
		2	2	Closed
		1	3	Closed
		1	4	Closed
19-030	19.4.2	1	1	Closed
?	19.4.3	2	1	Closed
			2	Open
			3	Closed
			4	Closed
			5	Open
19-031	19.4.3.1	1	1	Closed
			2	Closed

Track No.	PSAR Section	Round No.	Question No.	Status
19-032	19.4.3.2	1	1	Closed
			2	Closed
			3	Closed
			4	Closed
			5	Closed
			6	Closed
			7	Closed
19-033	19.4.3.3	1	1	Closed
			2	Open
19-034	19.4.3.4	1	1	Closed
			2	Closed
			3	Closed
19-035	19.4.3.6	1	1	Closed
			2	Closed
			3	Closed
19-036	19.4.3.7	1	1	Closed
			2	Closed
			3	Closed
19-037	19.4.3.8	1	1	Closed

Track No.	PSAR Section	Round No.	Question No.	Status
?	19.4.3.8	1	1	Closed
			2	Closed
			3	Closed
			4	Closed
			5	Closed
			6	Closed
			7	Closed
			8	Closed
			9	Closed
			10	Closed
			11	Closed

Lungmen PSAR Chapter 7
Open Issues

1. Ask TPC to provide more information on interlock capabilities, such as LTOP (reactor low temperature over-pressure protection) since the PSAR does not provide a lot of details in this area.

2. Ask TPC to provide justification for non-uniform SRV distributions on the remote shutdown panels.

3. Ask TPC to clarify why the turbine first-stage pressure RPS signal was not included.

APPENDIX 9A OPEN ITEMS

The criteria and approach used for the Lungmen Fire Hazards Analysis are acceptable. The FHA criteria and approach described in PSAR Section 9A conform to the applicable sections of BTP CMEB BTP 9.5-1 and the additional criteria from USNRC SECY 90-016, except for two open items. These open items are considered minor issues that do not need to be resolved prior to issuance of a construction permit but should be resolved prior to the issuance of an operating license. The two open items are:

1. The FHA needs to be prepared. This should be done in coordination with the completion of the design and the construction phase.

2. Smoke Migration into the RBSC HVAC System from External Hazards (SER Section 9A.1.4). Describe the RB secondary containment HVAC system design features that minimize the impact of smoke migration from external fires on all three safety divisions within the RB secondary containment.

3. Migration of Fire Suppressant (SER Section 9A.1.6). Provide the FHA approach to analyze the potential for and consequences of migration of fire suppressant (other than due to a pipe break) to other fire areas.

Evaluation of Open Items in Lungmen PSAR

Chapter 19 Review

Track No. 19-021

PSAR Section: 19.1

PSAR Question: 1

Round: 2

TPC has not made a commitment to include in Chapter 19 a deterministic assessment of severe accident prevention for significant risk contributors from external hazards. In particular, the seismic risk is the most dominant contributor for the Lungmen NPS (well over 90% of the total CDF). The USNRC policy statement on severe reactor accidents regarding future design and existing plants indicates that consideration of the dominant contributors to the severe accident risk and the insights identified by the PRA is necessary in addressing severe accident concerns for the assurance of no undue risk to public health and safety. TPC argues that they do not want to duplicate the same information in Chapter 3 and Appendix A to minimize inconsistencies and mistakes. This line of argument is invalid for the single most important risk contributor to the occurrence of severe accidents. If this position were to be taken, there would be no need for Chapter 19 at all since every subject discussed in Chapter 19 has been presented in other sections of the PSAR. In fact, as TPC indicated in the response to one of the review questions, the entire PSAR is an account of severe accident prevention and mitigation. The technical issues identified in SECY-90-016 were developed for the certification of generic US designs. USNRC indicated to us in phone conversation that they have not yet developed requirements and regulations for severe accident analysis for the application of construction permit or operating license because there has been no order of new plants in the US. It is implied that the technical issues specified in SECY-90-016 will not be the only ones required if there is such an application.

The probabilistic measure of the plant's ability to prevent and mitigate severe accidents (e.g., reliability of safety functions and conditional likelihood of core damage for various accident initiators) is necessary for the evaluation of safety margin and balance in the plant design for the various safety functions. The redundancy, independence, and diversity requirements cannot completely assure the safety margin and balance of design if all of the redundant and diverse systems have very poor performance. The evaluation of safety margin is specified in USNRC's draft Standard Review Plan for "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance" dated January 1998. Appendix A to the Lungmen PSAR does not include any intermediate results in this respect. In addition, only limited results (top 100 core damage cutsets) are presented in Appendix A. It is not clear what percentage of the CDF is contributed from these top 100 cutsets (perhaps a small fraction of the total CDF). This makes it very difficult to derive the kind of measures discussed in this question. TPC made no commitment to provide this information. This is a documentation issue.

TPC only made a commitment to include a summary of differences between the Lungmen NPS final design and the Standard ABWR that are relevant to severe accident prevention and mitigation. TPC made no commitment to provide a qualitative discussion of the effects of each of these differences on severe accident prevention and mitigation. This is just a documentation issue.

Track No.

PSAR Section: 19.3

PSAR Question: 1

Round: 2

TPC only provided a list of the major differences between the Lungmen NPS and the Standard ABWR. This list did not include the difference in fuel design. TPC never provided an evaluation of the effects of each of the differences on severe accident prevention and mitigation. This is just a documentation issue.

Track No. 19-024

PSAR Section: 19.3.1

PSAR Question: 1

Round: 1 and 2

Unresolved Safety Issue A-45 is considered resolved only if an IPE/PRA is completed and has demonstrated in the IPE/PRA that there are no vulnerabilities related to decay heat removal (DHR) functional failure. To demonstrate that there is no vulnerability related to decay heat removal functional failure, one has to show that the total contribution to the occurrence of severe accidents from the decay heat removal functional failure is acceptably low. This is done in every IPE program submitted to USNRC for the fulfillment of Generic Letter 88-20 requirements. In each of these submittals, a separate section is prepared to show that the total contribution to core damage risk from the decay heat removal functional failure is acceptably low. In the Lungmen PSAR, no such section exists. Furthermore, Appendix A to the Lungmen PSAR only presents the top 100 cutsets (which contributes to, perhaps, a small fraction of the total CDF). This can only show that there is no single cutset that has unacceptably high contribution. It does not show that the total DHR failure contribution is acceptably low, except that it cannot be higher than the total CDF minus the non-DHR failures in the top 100 cutsets. Since the USIs and GSIs are presented in Appendices to Chapter 1, this item is outside the scope of Chapter 19. This is only a documentation issue.

Track No. 19-024

PSAR Section: 19.3.1

PSAR Question: 2

Round: 1 and 2

See comments on Question 1 of Round 2 on Section 19.1 (Track No. 19-021).

Track No. 19-024

PSAR Section: 19.3.1

PSAR Question: 3

Round: 1 and 2

See comments on Question 1 of Round 2 on Section 19.1 (Track No. 19-021).

Track No. 19-025

PSAR Section: 19.3.1.1

PSAR Question: 2

Round: 2

TPC's response to this question provides a lot of information related to the criteria and procedural guidance for terminating all injection sources and dropping the water level to top of active to reduce power. However, it never addresses the question related to the control of low-pressure cold water injection after vessel depressurization, which is the main thrust of this question. However, this is not a minor issue. It will be no worse than the existing plants ~~and~~ it will not be a problem even if ROC-AEC wants to close this item.

Track No. 19-026

PSAR Section: 19.3.1.2

PSAR Question: 2

Round: 2

No TPC response has been received. This is not a major issue. It should be no worse than the existing plants. It will not be a problem even if ROC-AEC wants to close this item.

Track No. 19-027

PSAR Section: 19.3.1.3

PSAR Question: 1

Round: 2

No TPC response has been received. This is not a major issue. It should be no worse than the existing plants. It will not be a problem even if ROC-AEC wants to close this item.

Track No. 19-027

PSAR Section: 19.3.1.3

PSAR Question: 2

Round: 2

No TPC response has been received. This is not a major issue. It should be no worse than the existing plants. It will not be a problem even if ROC-AEC wants to close this item.

Track No. 19-027

PSAR Section: 19.3.1.3

PSAR Question: 3

Round: 2

TPC response to this question never addresses the question related to the main control room being a shared area between the two adjacent units. This is not a major issue. Most of the existing plants with dual units have shared main control room. It will not be a problem even if ROC-AEC wants to close this item.

Track No. 19-028

PSAR Section: 19.3.1.4

PSAR Question: 1

Round: 1 and 2

Based on TPC's response to the Round 2 question, it appears that the three alternative protection features will not be provided to any of the low pressure systems that interface with the RCPB and are not designed to the URS. The SECY-90-016 criteria specify that these three features should be provided to the low-pressure systems that are not designed to the URS. Based on the Lungmen PSAR, the low pressure sinks listed in Section 3M.3, heat exchangers listed in Section 3M.7, and low pressure lines not listed in Section 3MA are the only low pressure system sections that interface with the RCPB and will not be designed to the URS. TPC's response indicates that the three alternative protection features will not be provided to any of the above low-pressure systems. This seems to be inconsistent with the SECY-90-016 criteria.

Track No. 19-029

PSAR Section: 19.4.1

PSAR Question: 1

Round: 1 and 2

It is not clear if the access tunnels provide a pathway for debris dispersal during HPME. TPC's response to questions raised regarding the role of these tunnels in the progression of severe accidents does not appear to address the HPME scenario. TPC's response indicates that these tunnels are located at an elevation much higher than the largest possible debris depth. As such, debris accumulation in the tunnels is not an issue. However, the debris can be blown into the tunnels during a HPME. The potential for debris accumulation in the tunnels, the potential for melt-through, and the possibility of suppression pool bypass must be addressed. This is an issue that must be resolved.

Track No.

PSAR Section: 19.4.3

PSAR Question: 2

Round: 2

TPC's response to this question indicates that the following two reasons contribute to the higher fraction of zirconium oxidized for the base case in the low-pressure events than that in the high-pressure sequences:

1. Longer time to core uncovery for the high-pressure base case.

2. Slower metal/water reaction rate for the high-pressure case than the low-pressure case.

The first reason seems to be odd. Zirconium oxidation and hydrogen generation occur after the core uncovery. Time to core uncovery does not appear to be a factor. In fact, there should be more time available to react with steam for the high-pressure case. This is not a major issue. It will not be a problem even if ROC-AEC wants to close this item.

Track No.

PSAR Section: 19.4.3

PSAR Question: 5

Round: 2

TPC's response never states that the approach used is identical to that used in NUREG/CR-6338. It was indicated by TPC that a similar approach was used. TPC did not provide a description of the exact calculation process as requested. This is a suspect. It is possible that the probability of containment failure was not obtained by calculating area lying under the intersection of the two probability density curves (stress and strength). Numerically, it may not make a large difference. Mathematically, the calculation method is probably incorrect. ROC-AEC may want to double-check the calculation. In addition, as indicated in the TPC response, a truncation was applied for pressures lower than the level C pressure of 0.77 Mpa. It makes sense to truncate, as

long as it is less than the test pressure, which is true normally. This is not a major issue. It will not be a problem even if ROC-AEC wants to close this item.

Track No.

PSAR Section: 19.4.3.3

PSAR Question: 2

Round: 1

See comments on Question 5 of Round 2 on Section 19.4.3 (no Track No.).

ADDENDUM 2 TO MPR SER ON PARTIAL REVIEW OF LUNGMEN PSAR

Recommendations for Future Actions

Recommendations Based On Review Of The Lungmen PSAR Sections 3.7 And 3.8 And Chapter 19

The following recommendations have been developed based on the insights obtained from the review of the Lungmen PSAR Sections 3.7 and 3.8 and Chapter 19.

General Recommendations

1. All outstanding items be resolved and proper commitments be made by TPC, as appropriate, in resolving these issues.

2. Tracking system be established for all open items and periodic verification should be performed to ensure that all TPC commitments are met appropriately.

3. Periodic review be performed to ensure adherence to plant design features specified in the PSAR for seismic system and subsystem analyses, design of seismic Category I structures, and severe accident prevention and mitigation.

Sections 3.7 and 3.8

1. Sections 3.7 and 3.8 of the Lungmen PSAR referenced other sections of the PSAR in different subsections. For example:

   - Subsection 3.7.3.1 Seismic Analysis Methods - PSAR Sections 3.9.2.2 and 3.10 were referred to for equipment testing requirements.

   - Subsection 3.7.3.8.9 Design of Small Branch and Small Bore Piping - PSAR Section 3.9.3.4 was referenced for design of ASME III instrumentation lines supports.

- Subsection 3.8.1.3.3 Severe Environmental Loads - PSAR Section 3.3 was referenced for design wind and typhoon loads.

- Subsection 3.8.4.3.1.1:Loads and Notations - Subsection 3.5.1.4 was referenced for typhoon generated missiles.

There are instances that TPC responded to the PSAR review questions by referring to criteria presented in other PSAR sections (e.g., Section 3.9 of PSAR) which are out of the review scope. The reviewers have accepted such responses without performing a review of these out-of-scope PSAR sections. It is recommended that PSAR Sections 3.3, 3.5, and 3.9 be reviewed parallel with Sections 3.7 and 3.8 for completeness and consistency.

2. Since the GE ABWR was designed for a generic 0.3g SSE and the Lungmen SSE is 0.4g, it is recommended to perform an in-depth review of TPC's commitments to seismic qualification of safety-related mechanical and electrical equipment.

3. A coupled dynamic model consisting of the containment structure, containment internal structures, reactor building, as well as the RPV and reactor internals (PSAR Subsection 3.7.2.3.2) will be used for performing seismic response analyses. The PSAR did not provide details of the coupled model. However, the model is expected to be complex. Furthermore, the model will be used in the frequency dependent soil-structure interaction analysis using program SASSI. It is recommended that both the coupled model, modal analysis results of the model, and the SASSI analyses be reviewed when available.

4. Identify major items of SSC for performing audits on structural design calculations or design reports.

5. Perform a review on the development of hydrodynamic loads resulted from SRV actuation and LOCA and their effects on both systems and subsystems response.

Chapter 19

1. A Standard Review Plan should be developed for FSAR Chapter 19 to clearly define the scope, content, acceptance criteria, etc.

2. The possibility of debris accumulation in the access tunnels during the HPME scenario is an important issue that affects the plant design. TPC's response to questions raised during the PSAR does not address the potential for debris accumulation in the tunnels due to HPME, the potential for melt-through, and the possibility of suppression pool bypass. This issue must be properly resolved to ensure that there is no vulnerability in plant design for severe accident mitigation.

3. It is suggested that ROC-AEC request TPC to provide a copy of the models developed for the preparation of the Lungmen PSAR so that ROC-AEC can perform independent calculations to check/verify information provided by TPC or to perform sensitivity evaluation.

4. Section 19.5 of the Lungmen PSAR lists the features of the Lungmen containment that mitigate severe accident phenomena and contribute to limiting the conditional containment failure probability (CCFP). However, it does not identify a quantitative containment performance goal that is being used as a benchmark nor does it state that this goal has been achieved. It should be ensured that these important aspects of containment performance be addressed in the FSAR.

Lungmen PSAR Chapter 7 Recommendations

1. An inspection and testing plan for the I&C should be prepared to ensure that I&C systems, particularly digital systems, are built in accordance with the PSAR. The plan should include:

   - Definition of inspections to be made
   - Definition of tests to be applied
   - Times during the I&C development and implementation when inspections and tests will be applied

2. Inspections and tests should focus on key systems features including:

   - Integrated system timing and loading, e.g., stress tests
   - Software quality, e.g., completeness of V&V and documentation
   - Human factors

3. Inspections and tests should be carried out throughout the development and implementation of the digital systems hardware and software

Note: Attached are answers to ROC-AEC comments sent on June 11, 1998. These include some recommended follow-up actions.

Response to ROC-AEC June 11, 1998 Comments on Chapter 7 - I&C Systems

Question 1:

Per the description of Lungmen PSAR prepared by TPC, the remote S/D system (RSS) belongs to analog system, and it interacts with the digital main control room system as diversity. Therefore, there is no digital regulatory compliance in this particular system. Yet, in the MPR question of 07-017 response prepared by TPC, it stated otherwise. It responded with a possibility to facilitate the microprocessor-based components in the future. This will certainly contradict the description stated before. MPR did not express any opinion on this issue. We wonder whether this is of no concern to MPR at all. Does this mean the above is alright to accept?

Answer:

MPR considers the use of diverse microprocessors in the remote shutdown panel to be acceptable. The use of diverse digital components in the shutdown panel adequately addresses the concern that the reactor shutdown panel be diverse from the main control room systems, and alleviates the need for the panel to be all analog. We understand that the USNRC Staff agrees with that approach.

Question 2:

PSAR Sections 7.5 & 7.6 are not in compliance with the SRP requirements, e.g., the requirements of important safety parameter display system, SPDS in Section 7.5, and the interlock of the reactor low temperature high pressure of Section 7.6 both were not presented in the PSAR. This part became a review question and has been sent to TPC for response. What is MPR's opinion on this topic?

Answer:

SPDS monitoring capabilities are integrated in the normal and diverse instrumentation and not broken out as a separate system. TPC has committed to NUREG 0737 compliance, so the SPDS capabilities ultimately provided should be satisfactory (compliance will of course have to be checked later when more details are available). We suggest TPC be asked to provide more information on interlock capabilities, such as LTOP, since the PSAR does not provide a lot of details in this area. However, the commitments in the PSAR for compliance appear adequate. We consider additional information would be useful for ROC-AEC to obtain but it

is not a prerequisite to acceptance of the PSAR and issuing the construction permit (CP).

Question 3:

There are two remote shutdown panels with non-uniform SRV distributions. A total of three SRVs in the system, one panel has to equip with two SRVs, and the other panel has only one. Will this affect the panel capability of unique shutdown?

Answer:

The PSAR describes a remote shutdown capability that meets regulatory requirements. The remote shutdown capability is not required to provide all safety controls nor be fully redundant. However, we suggest TPC be asked to provide further details on this to help address your question. The additional details would be useful for ROC-AEC to obtain but are not a prerequisite to acceptance of the PSAR and issuing the construction permit.

Question 4:

TPC has committed to comply with the IEEE 279 for the RPS on Lungmen Project I&C systems. But, the staff has recognized that IEEE 279 deals mainly analog system and has been in application for long time and has been replaced by IEEE 603. The IEEE 603 has also been endorsed by the U.S. NRC R.G. 1.153. And, the Lungmen PSAR has a commitment to comply with R.G. 1.153. Though TPC did not explicitly say that they will adopt IEEE 603, they are indeed using this new standard indirectly. Is it true that IEEE 603 can supersede IEEE 279 in the Lungmen Project application? We would like to emphasize that Lungmen Project is a brand new NPP, and with a brand new license application. The status of the license is certainly different from the existing licenses.

Answer:

As discussed at the meeting, MPR agrees that the Lungmen design should follow IEEE 603. Further, the PSAR, see Table 7.1-2, commits to IEEE 279, R.G. 1.153 and IEEE 603. Accordingly, we consider the commitments in this area to be complete. In answer to your question, IEEE 603 can supersede IEEE 279 for the Lungmen Project. As you probably know, the NRC is in the process of changing the U.S. regulations to replace IEEE 279 with IEEE 603. How IEEE 603 is imposed officially by the ROC-AEC is an internal matter to be resolved by ROC-AEC.

Question 5:

Per MPR Question 07-013 on software developments, TPC has responded to the question and sent partial software development plans:

(1) Software Management Plan;
(2) Software Configuration Management Plan;
(3) Software Verification and Validation Plan;
(4) Software Development Project Plan.

These development plans are the key documents for the system installation, and are considered as very critical to this project. What is MPR's general opinion on these documents?

Answer:

These documents are considered to be sufficiently thorough and satisfactory to guide the Lungmen software development. However, we recommend that ROC-AEC confirm that the actual digital systems are developed in accordance with these plans. Rigorous inspections and testing by ROC-AEC are recommended to confirm the guidance is followed, and appropriate actions are taken in response to findings from the reviews and testing required by the plans. The acceptance criteria the ROC-AEC should use would be those stated in the regulations and SRP and are not constrained by what is stated in the software development plans.

Question 6:

The scram signal of Lungmen Project has considered some plant specific features which differ from the standard GE ABWR. However, the GE ABWR has the design feature of mainsteam line with high radiation scram signal that Lungmen Project does not have. This situation has been raised by MPR question of 07-016. GE responded to this question with a GE report: NEDO-31400A. It explains that the removal of that signal can avoid the malfunction of the instrument to trigger the reactor scram. The ROC-AEC staff thinks this problem can be resolved by enhancing the reliability of the instrument to avoid the possibility of malfunction of the instrumentation, and not the total removal of the system. Even if the removal of the system, an alternative design should be added to the system to compensate the situation.

Answer:

Although we have not been provided NEDO-31400A for review, we understand that it justifies the removal of the subject scram by reaching acceptable conclusions on accident analyses without the need for the mainsteam line high radiation scram. In general, we consider it prudent to remove any scram that is shown by appropriate

accident analysis to be unnecessary. Although we agree that key instrumentation can be made highly reliable, there will always be the possibility of a malfunction due to one cause or another (e.g., a maintenance or testing error). Because spurious scrams represent challenges to the plan safety systems, we consider removal of unneeded scrams a better approach.

Question 7: The Sizewell-B NPP has experienced installation failure of digital I&C at the early plant construction stage in England. It took big effort and change of management to fix the problem. We wonder that the similar situation could occur to the Lungmen Project since this would be our first NPP with the adoption of new digital I&C systems for the plant. We already knew that the detailed design and implementation of the digital I&C at Lungmen will not be the same as that of K6/K7 in Japan. It will be a real challenge for us to proceed the project in this area. Please elaborate the precautions and procedures that have been established for those subcontractors to ensure a success installation of the digital I&C. As of now, are they in better shape of the advanced technologies, or getting more experienced to avoid the repetition of the failure of Sizewell-B?

Answer:

There is an increased recognition of the importance of addressing system aspects in the design of digital systems to preclude the type of problem that occurred at Sizewell-B. The revised USNRC SRP provides guidance in this area and the vendors are more experienced and aware then they were previously. We consider this concern should not prevent going ahead with the project.

We agree it will be a major challenge. The competence and experience of people and organizations involved will be critical factors in success. The NRC in particular looks at completeness and experience in their reviews. We recommend that ROC-AEC institute a rigorous set of inspections and tests of the digital systems, and organizations and process used, throughout their development and installation. This will be the best way to minimize the chances of digital I&C failures or inadequacies. Specific checks recommended are provided in the cover sheet for this attachment.

Question 8:

In GE ABWR SSAR Table 7.2-1, one of the RPS signal is turbine first-stage pressure. It is of interest to now why this signal has been removed from Lungmen PSAR?

Answer:

The first paragraph of Section 7.2.1.1.4.2 of the GE ABWR SSAR list of reactor scram initiating conditions does not list turbine first-stage pressure. Accordingly, the Lungmen PSAR may have omitted the turbine first-stage pressure signal because of this. We suggest ROC-AEC request TPC to clarify this issue. As noted above, we prefer the simplest system able to protect the plant so we would not add this trip without a clear reason to do so.

Question 9:

Many important system responses are from non-safety systems such as RFCS and RCIS. Earthquake may be a source of common cause failure. Given an earthquake event, it is of interest to know whether the digital equipment is more reliable than the analog equipment or not.

Answer:

Safety systems are required to be designed and qualified to withstand natural phenomena challenges, such as earthquakes. IEEE 279 and IEEE 603 compliance requires adequate seismically qualified isolation of non-safety systems from safety systems which will prevent potential problems where non-safety systems are connected to safety systems. Also, digital systems tend to be very rugged seismically. Further, TPC is committed to look for such weaknesses in the I&C design and if they find them to fix them. We do not consider this to be a basis for concern about the use of digital systems, provided they are seismically qualified.

RECOMMENDATIONS FOR FUTURE ACTIVITIES FOR LUNGMEN PROJECT FIRE HAZARDS ANALYSIS REVIEW

The Fire Hazards Analysis (FHA) had not been performed for the Lungmen PSAR submittal to ROC-AEC. The FHA will be performed during plant design and construction using the criteria and approach documented in the PSAR. The FHA will be referenced in the FSAR.

The primary future activity for ROC-AEC regarding the FHA review is the review of the formal FHA. Because the FHA documents the design basis of the fire suppression and detection systems, as well as the safe shutdown capability of the plant, ROC-AEC should audit the development of the FHA, implementation of fire hazard design requirements, and implementation of TPC commitments regarding the FHA on a regular basis to assure compliance to the FHA criteria.

RESPONSES TO ROC-AEC COMMENTS FROM JUNE 8-9, 1998 MEETING

Review Of Unit 2 Construction Fire Hazards During Unit 1 Operation

TPC committed to having the Site Fire Protection Program in place and to address actions in case of fires on the Unit 2 construction site. TPC also committed to performing an engineering review of temporary electrical or mechanical system cross connects to unit 2 as part of the Unit 1 completion program. This engineering review would verify that any cross-connects do not invalidate the Unit 1 Fire Hazards Analysis or Safe Shutdown Analysis. In ROC-AEC Facsimile from R. T. Sun to D. A. Freed (MPR) dated June 11, 1998, MPR was requested to comment on the key areas to be aware of and major concerns that the program should include (Comment 1 on Appendix 9A). The following key elements should be addressed by the site fire protection program and Unit 1 completion program to ensure adequate fire protection and safe shutdown capability at Unit 1:

- The availability of the fire water supply and fire main loop serving Unit 1. A loop should be provided for Unit 1 at all times.
- Protection of the fire main loop from damage due to Unit 2 construction activities.
- Isolation of construction related combustible loads, explosion hazards, and ignition sources from the Unit 1 buildings and systems. Isolation may be achieved by physical barriers or administrative controls.
- Temporary electrical and mechanical system ties between Unit 1 and the construction site and their impact on the FHA and safe shutdown analysis.
- Protection of facilities shared between the two units (e.g., swing diesel generator, RBSW building) during Unit 2 construction. In particular, these facilities should be specifically addressed by the program if construction activities are occurring in them during Unit 1 operation.

- Temporary and permanent fire protection features (barriers, detection systems, and suppression) for the construction site to assure quick detection and suppression of any fires that may occur.
- Potential exposure fire, heat, smoke, or water exposure on Unit 1 resulting from construction site fires.

Potential for Multiple, Simultaneous Fires

In ROC-AEC Facsimile from R. T. Sun to D. A. Freed (MPR) dated June 11, 1998, MPR was also requested to discuss the implications of Information Notice 97-01 regarding inadequate grounding resulted in simultaneous fires in more than one fire area (Comment 2 on Appendix 9A). The situation in the information notice consisted of an isolation transformer for an emergency lighting uninterruptible power supply panel shorting to ground. The neutral leg of the transformer had not been grounded. Instead, the neutral leg of an inverter that also supplied power to the emergency lighting uninterruptible power supply panel had been grounded. The short in the transformer resulted in excessive current in the panel and the power supply with the inverter, causing circuits in these two locations to ignite. The root cause of this fire was attributed to improper grounding due to an electrical design error. There is no fire protection regulatory requirement related to the situation in the information notice. Instead, IEEE standards, such as IEEE-142, provide guidance on grounding. It is believed that use of the standards should result in a fault tolerant electrical system design.

There are fire protection criteria related to isolation of associated circuits in 10CFR50 Appendix R and BTP CMEB 9.5-1 to assure a hot short, short to ground, or open circuits of an associated circuit will not prevent operation of redundant or alternative safe shutdown equipment. Clarification of the definition of an associated circuit and guidance to protect against them are provided in Generic Letter 81-12 and an April 7, 1982 clarification to Generic Letter 81-12. These criteria may be used to evaluate electrical system grounding design criteria, although the subject circuits in the Information Notice were not necessarily associated circuits. The foreseeable risk which may cause fire in more than one fire zone are inadequate fire barriers (including seals, fire doors, and fire dampers) and excessive fire loading. Proper electrical system design to current standards should not result in fires in more than one fire zone.

Comments on Completeness and Soundness of TPC's Commitment for FHA

In ROC-AEC Facsimile from R. T. Sun to D. A. Freed (MPR) dated June 11, 1998, MPR was requested to comment on the completeness and soundness of the TPC commitment in performing the design and analysis work of the Fire Hazards Analysis section (Comment 3 on Appendix 9A). The TPC approach for the design and analysis of protection against fire hazards as documented in the PSAR and the TPC responses to questions generated during the Safety Evaluation Report review appears complete and appropriate. Plant layout and the use of three safety divisions provide more inherent fire protection and safe shutdown

protection than current operating plants. MPR considers the following to be the key criteria to monitor during the design and analysis work:

1. Identification and quantification of all combustibles and ignition sources in a fire area, including in situ and transient. This information determines the fire loading and potential for ignition for the fire area.

2. Continuity of the combustibles in the fire area. This information determines the distribution of the fire loading.

3. Continuity and qualification of fire barriers including doors, dampers, electrical and mechanical penetrations, structural steel protection, and seismic gaps. This information determines the capability of the structure to contain the fire.

4. Fire detector placement and coverage. This information determines the capability of fires to be detected early.

5. Fire suppression system placement, coverage, and actuation method. This information determines the capability of suppressing any fires that occur.

6. Protection of safe shutdown equipment from inadvertent operation of fire suppression systems and from potential seismic interactions. This information determines the potential for damage to safety related electrical components resulting from inadvertent spraying or fire suppression system pipe leaks or breaks during a seismic event.

7. Routing of divisional cables for safe shutdown systems to ensure appropriate separation (per Appendix R, Section III.G.1.a). This information determines the extent to which the safety divisions are separated by fire barriers.

8. Protection of associated circuits as defined in BTP CMEB 9.5-1, Position C.5.b. A clarification of the definition for associated circuits is provided in Generic Letter 81-12 and a April 7, 1982 clarification to Generic Letter 81-12. This information determines the extent to which the consequences of a fire may propagate beyond the fire area with the fire. The consequences include propagation of the fire along the associated circuit or propagation of damage to other safety divisions such as tripping of power supplies of redundant shutdown equipment or spurious actuation of components that may impact safe shutdown.

9. Shutdown from the Remote Shutdown Panels (RSPs) as a result of a fire in the Control Room Complex, including independence and separation of circuits for the RSPs from Control Room Complex circuits. This information determines the ability to safely shutdown the plant using alternative shutdown capability.

10. Protection of shared facilities from fires that could affect both units. This information determines the potential for a fire challenging safe shutdown at both units.

11. Protection against external fire hazards, including smoke migration. This information determines the potential for external fires to challenge safe shutdown.

12. Potential for migration of smoke, hot gases or fire suppressant into other fire areas. This information determines the potential for the spread of the fire or damage to shutdown equipment in other fire areas.

The focus of the ROC-AEC reviews using the above criteria should be on fire areas containing safe shutdown equipment.

FUTURE ACTIVITY RECOMMENDATIONS FROM SER

The SER review identified several items that should be monitored during the design and construction of the plant. These items are discussed below.

1. **Review Administrative Procedures Controlling Transient Combustibles and Ignition Sources.** These procedures will be developed as part of the overall fire protection program when the FSAR is being produced. The final FHA should identify areas where transient combustibles are limited and controlled. The administrative procedures should be consistent with the final FHA.

2. **Review FHA to assure Reactor Building secondary containment considers the potential for smoke migration from the fire area to other fire areas of the same division.** The secondary containment HVAC smoke removal mode assures a pressure differential between fire areas of different divisions but not between fire areas of the same division. TPC committed to evaluate this condition.

3. **Verify FHA identifies areas of the plant with explosion hazards to safety related equipment.** No areas with explosive hazards have been identified yet, but TPC committed to identifying and providing appropriate protection against explosion hazards.

4. **Verify FHA identifies any design provisions to protect safety related equipment from inadvertent operation of fire suppression systems.** Automatic suppression systems may be added during the design process. These systems should not result in damage to safety related equipment if inadvertently actuated. Of particular concern is the potential to spray down electrical cabinets to cause shorting and the protection against this.

5. **Verify FHA demonstrates that multiple high impedance faults will not negatively impact other equipment fed from the same power source.** TPC committed to this in PSAR Section 9.5.1.4.

6. Verify FHA includes all buildings and sites containing safety related equipment and that these buildings receive the same level of protection as the Reactor Building,

Control Building, and Turbine Building. TPC indicated that the Fire pump house, Auxiliary Building, Fuel Building, Switchgear Building, RBSW pump house, Access Control Building, Switchyard, and other yard structures will be included in the FHA. Of particular concern are the RBSW pump house containing RBSW pumps for both units, yard structures and other buildings that may pose exposure hazards to safety related equipment, and other buildings with facilities shared between the two units.

7. Verify FHA considers the potential for smoke borne radioactive releases. TPC indicated that the combustible load data for fire areas will be compared to radiation zone drawings to identify potential areas for radioactive releases.

8. Verify fire barriers, including walls, doors, penetrations, dampers, structural steel fireproofing are rated appropriately, or are designed, tested, and tested to UL standards. In particular, PSAR 9A.3.5 identifies wall designs that may require UL testing or testing per UL standards to verify fire rating, and PSAR 9A.3.6 identifies fire doors that may not be labeled as fire doors. For both of these situations, the fire resistive rating of the barrier needs to be demonstrated and documented.

9. Verify the basis for the justification for not installing fire detectors in the Control Room Complex consoles and cabinets as stated in the PSAR and verify the FHA addresses this basis. BTP CMEB 9.5-1 recommends fire detectors inside consoles and cabinets in the Control Room Complex. PSAR Section 9.5.1 provides justification for not installing the fire detectors. The bases for this justification need to be verified. Further, the FHA should verify the bases.

10. Verify fire suppression requirements from BTP CMEB 9.5-1 are met or that deviations are identified and provided with adequate justification. An example of a potential deviation from the BTP is provided in the SER regarding the need for automatic suppression for cable tray stacks containing safety related cables.

11. Verify electrical independence of Remote Shutdown Panels from Control Room Complex. This independence should assure that no associated circuits may disable control of safe shutdown equipment from the Remote Shutdown Panels as a result of a fire in the Control Room.

12. Verify the electrical configuration for the swing diesel generator precludes the potential for disabling electrical systems at either or both units as a result of a fire. The swing diesel generator was identified as a system shared between the two units that interfaces with the electrical system of each unit. The potential of damaging electrical equipment, including tripping of electrical buses, at either or both units needs to be assessed as a result of a fire involving the swing diesel or its components.

13. Verify that the design and operation of the Reactor Building secondary containment HVAC minimizes the potential to distribute smoke to all safety divisions inside secondary containment as a result of an external fire. This HVAC system is unique in

that one supply system and one exhaust system serve all three safety divisions. Information on the design features to minimize this common mode scenario was requested from TPC in the SER and is an SER open item.

14. **Verify FHA addresses migration of fire suppressant to other fire areas, particularly for areas that would be subject to manual fire suppression.** The FHA appears to address only areas with fire protection piping that may rupture or has automatic suppression systems that may inadvertently actuate.

15. **Verify the validity of the justifications for the special cases listed in the PSAR** Section 9A.5. The FHA should verify the bases for the justifications of these special cases. Of particular concern are the hazards related to charcoal filters for process tanks and the areas with circuits of more than one safety division.