ENCLOSURE 2

PERRY INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(FRONT-END)

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Perry Nuclear Power Plant Unit 1 IPE: Step 1 Front-End Audit

Contractor Step 1 Audit Report NRC-04-91-066, Task 6

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I. INTRODUCTION

This introductory chapter presents the process used by Science and Engineering Associates, Inc. (SEA) to audit the front-end portion of the Cleveland Electric Illuminating (CEI's) Individual Plant Examination (IPE) Submittal for the Perry Nuclear Power Plant (PNPP) Unit 1. This front end review focuses on accident sequences leading to core damage, due to internal initiating events and internal flooding. Audits of the human factors analysis and back-end analysis were performed by the NRC with contractual help from Concord Associates, Inc. and Scientech, Inc., respectively. There have been discussions between these teams to check IPE treatment of Level 1/Level 2 interfaces, and Level 1/Human Factors interfaces. The contractor review findings are presented in Section II, and IPE Evaluations and Data Summary Sheets are enclosed as Section IV.

I.1 SEA Audit Process

This audit is a Step 1 audit, which means that the issues raised in this report have not been discussed with the CEI personnel. Also, a visit to the PNPP site is out of scope of this audit. The purpose of this audit is to identify issues related to the front-end IPE analyses for PNPP Unit 1, and to provide NRC with these issues. SEA Audit Process is illustrated in Figure 1 and subsequently described below.

I.1.1 Review of FSAR and Tech Specs

The NRC provided the PNPP submittal to SEA in September 1992. SEA began work on September 11. Between September 11 and 30, the review focused on a horizontal review of the submittal to develop sufficient understanding of various front-line and support systems, and identify apparent deficiencies, if any, in the information assembly process of the IPE. The objective of the preliminary review was to identify specific areas in the FSAR that need special attention.

Between October 4 and 30, the latest (updated) Final Safety Analysis Report (FSAR) and Technical Specifications (Tech Specs) for PNPP were reviewed. Copies of the required parts of the FSAR and TechSpecs were brought to SEA, Albuquerque with permission of the NRR Project Manager, Mr. J. R. Hall. This provided additional time for FSAR review as well as permitting the FSAR to be referred to during various stages of IPE review. The focus of the review was to obtain a better understanding of various plant systems, of plant design, and of accident response, and to determine whether the licensee modeled the as-built and as-operated plant.

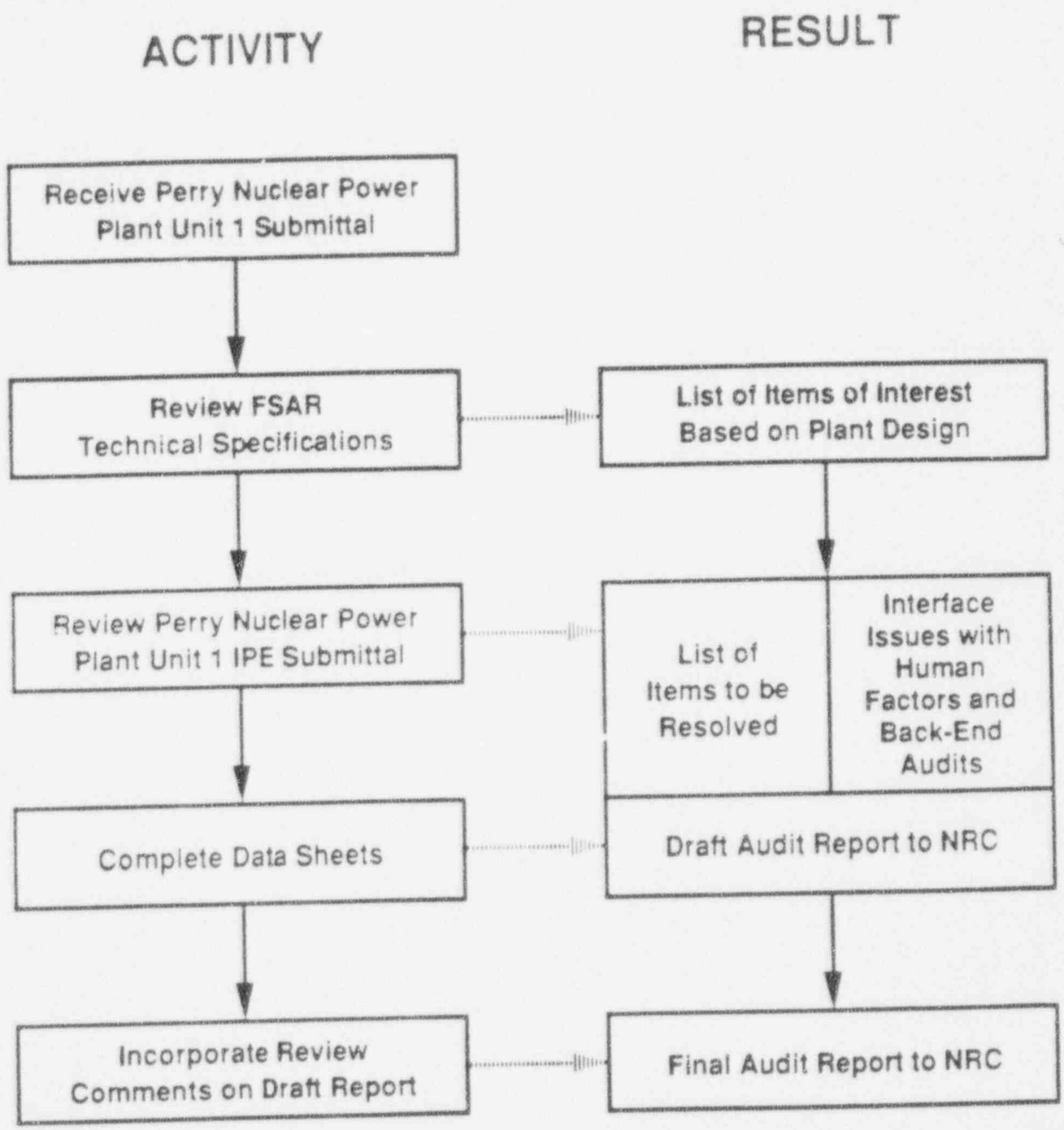

Figure 1. SEA Step 1 Audit for Perry Nuclear Power Plant Unit 1 Front-End IPE

1.1.2 Review of IPE Submittal

Between October 4 and October 30 a detailed review of the IPE submittal for PNPP was accomplished. The effort incorporated a horizontal review of all aspects of the front-end issues called for in Step 1 Review Guidance Document (dated 05/19/92), as well as vertical reviews of selected key issues. The findings of the review are documented in Section II of this report. The review procedure focused on checking each check-item listed in the Step 1 Review Guidance Document as well as the Statement-Of-Work (SOW).

1.1.3 Audit Report

On November 6, a draft copy of the audit report was sent to NRC for review. The report addressed each work requirement called-out in the SOW. The report also includes (Section IV) a set of IPE Evaluation and Data Summary Sheets. The standard format for these summary sheets was provided by the NRC. These sheets were completed as required.

The final report will incorporate the comments from the NRC on this Draft report.

1.2 PNPP IPE Methodology

The PNPP IPE uses the small event tree and large linked fault tree methodology to perform the front end analyses. Detailed fault trees were developed to the component and train level, for each of the front-line and support systems. Recovery actions were considered and common mode failures were incorporated into the fault trees. Generic and Plant-specific failure data were used in the analysis. NUPRA was used for sequence quantification and fault tree linking. Data uncertainty analysis was performed, also using NUPRA, to quantify data uncertainty and its impact on the CDF.

The methodology used in the IPE front-end analysis of PNPP Unit 1 meets the criteria of NUREG-1335 and Generic Letter 88-20.

1.3 PNPP Plant

The PNPP site consists of one BWR nuclear unit, BWR/6-238, manufactured by General Electric Company. The turbine generator was supplied by General Electric Company, and the engineering and construction was by Gilbert/Commonwealth. Unit 1 was declared commercial on November 1987 and is rated at 3579 MWt and 1250 MWe. Unit 2 is partially complete and is expected to be completed in the future. Some of the equipment in Unit 2, e.g., DC Batteries, are used at present for Unit 1 operation.

1.3.1 Similar Plants and PSA's

PNPP is similar to Grand Gulf nuclear power plant which is a BWR/6-251. This plant is a NUREG-4550 plant. Other similar plants include generic General Electric 238 Nuclear Island (GESSAR II), and Kuosheng which

is a BWR 6-218. PRA studies exist for these plants also although they are not [illegible] reviewed by the NRC. The differences between Grand Gulf and PNPP appear to be minor [FSAR Chapter 1] and none have shown to significantly impact the CDF outcome.

1.3.2 Unique Features

Unique features of PNPP Unit 1 include:

1. There is a motor driven feed pump that is normally in standby and will start on an automatic signal at Level 2, following failure of the turbine driven pumps.
2. The safety-related dc buses can be cross-tied to the dc batteries in the not yet completed Unit 2. This feature extends availability of dc power following LOOP and Station Blackout sequences.
3. The HPCS D.G. is not of same size or type as the other two D.G.s. This diversity will reduce the likelihood of some common cause failure of all three D.G.s.
4. The HPCS D.G. can be cross-tied to Division 2 emergency bus, which enables the containment vent valves and hydrogen ignitors to be powered in the event of LOOP and loss of Div. 1 and 2 D.G.s.
5. Containment failure leads to injection failure.
6. Makeup to the Suppression Pool is provided by gravity head. No pumps are involved.
7. HPCS and LPCS inject inside the core shroud. Consequently, the HPCS is not a recommended system for ATWS mitigation.
8. The "ADS inhibit" action is not automatic.

Based on the unique features, the key areas identified for review are ATWS and LOOP.

II. CONTRACTOR REVIEW FINDINGS

A team consisting of CEI personnel and contractor personnel has performed the IPE. The IPE was peer reviewed by CEI personnel with contractual help. It is stated that CEI intends to maintain the IPE as a living document with updates every two years. Main updates will include usage of more plant specific component failure data and human actions.

II.1 Review and Identification of Front-End Analysis

This section presents our findings, including a summary of IPE strengths and weaknesses and a list of questions to the licensee. The following sections address each work area explicitly in the order they appear in the SOW.

II.1.1 General Overview of Front-End Analysis

II.1.1.1 Completeness Check

Between September and October 7, a detailed review of the PNPP IPE submittal was accomplished. Initial review effort focused on a completeness check wherein the content of the IPE submittal was carefully examined to see if the information presented and the level of detail to which it was presented met the guidelines set by NUREG-1335. The PNPP IPE submittal closely adhered to the format recommended in NUREG-1335, which made this initial review process straightforward. **Based on the review it is concluded that the PNPP IPE submittal is complete with respect to the type of information and level of detail requested in NUREG-1335.** The Step 1 Review Guidance was extensively used in the review.

II.1.1.2 Methodology Check

The PNPP IPE uses Small Functional Event Tree and Large Linked Fault Tree methodology to perform the front-end analyses. It is reported that fault trees were developed for all front-line and support systems. Recovery actions were considered. Common mode failures were incorporated into the fault tree models. Uncertainty analyses were carried out. In addition, a sensitivity analysis and importance ranking were performed as part of the IPE effort. An internal flooding analysis was carried out to the level of detail required by NUREG-1335 and the GSI A-45 issue was adequately addressed. **In conclusion, the methodology used in the PNPP IPE submittal is consistent with the methods identified in Generic Letter 88-20, and NUREG-1335.**

II.1.1.3 Does IPE Model As-Built, As-Operated Plant

Section 2.4 of the IPE submittal discusses the information gathering process employed by the utility. According to the submittal, the description of each system was based on the most current versions of the FSAR, P&IDs, I&C drawings, and other related documents. For all the front-line systems, the systems description was

checked against USFAR as part of our review. To confirm that system models accurately represent the as-built as-operated plant, the following procedure was followed by the licensee.

1. The study was performed by the Independent Safety Engineering Section at the Perry site so that design documentation was directly available.
2. Analysis files were set up for each phase of the model development to ensure that the documents used and the decisions made on the basis of information in a given document were recorded. This ensures that comparison between the model and subsequent design change packages can be made in a controlled manner.
3. The design engineers reviewed all the system models for correctness of assumptions concerning design, alignment and operation.
4. Operations staff reviewed all the event trees.
5. The current set of operating procedures were used in performing the human reliability analysis and many of the actions were discussed with training and operations personnel.
6. Maintenance data was acquired directly from plant operating experience.
7. A significant number of visits were made to the plant to walkdown systems which could lead to flooding and to trace potential flood propagation pathways.
8. The Containment Building Strength Evaluation and portions of the internal flooding analysis were performed by Gilbert Commonwealth, the architect/engineer (A/E) for the Perry Nuclear Power Plant.
9. In addition to reviews of each of the system analyses by the appropriate design engineers, intermediate reviews of the work products and the draft report were performed by key personnel from the operations, training, and engineering departments.

This procedure appears to be thorough and ensures that plant models represent as-built, as-operated plant.

II.1.1.4 Internal Flooding Methodology

As part of this review, the IPE submittal Section 3.3.7 covering the internal flooding scenario was reviewed. Section 3.3.7.1 indicates that the full flooding analysis is described in Appendix G. However, Appendix G was not submitted for this review, and comments are based on observations documented in Section 3.3.7. The total contribution (from internal flooding) of ~1.5 x 10[illegible]/yr. represents a 12% contribution to core damage frequency from internal events and flooding. In Section 3.3.7, the licensee stated that a 12% contribution (with a largest single flooding scenario contributing about 7% of the 12%) is a conservative estimate and is acceptably small. As a result, no plant specific improvements were proposed by the licensee.

The following are specific comments/areas of concern that are either implicitly assumed and not documented, addressed in Appendix G, or were omitted from the flooding analysis.

II.1.1.5 Utility Peer-Review

An independent review team, consisting of several CEI engineers and contractor personnel from RAPA, conducted the peer-review. The CEI staff was entirely responsible for review of the systems models and the accident sequence models. Several engineers and licensed operators were trained initially in PRA methodologies and were then used in the review process. After the quantification process was complete, the corporate technical staff were responsible for review of the dominant sequences. The plant shift supervisor was the coordinator of this effort. In addition, two independent reviews by RAPA have taken place during the IPE effort. Phase I of the review covered initiating events, accident sequence analysis, and system modeling. Phase II covered common cause and dependency analysis, data base, human interactions, internal flooding, and sequence quantification. The major comments generated by the IPE peer review process were documented as part of the IPE (Section 5.3) and reflect thoroughness of the review process. These comments were adequately addressed in the IPE Section 5.4.

II.1.2 Review of Accident Sequence Delineation and Systems Analysis

II.1.2.1 Initiating Event Review

The review of the initiating events was carried out as recommended by Section 3.1.1 of the Draft Step 1 Review Guidance. The findings of the reviewer are as follows.

The identification of the Initiating Events (IEs) is based on standard techniques consisting of industry operating experience, other PRAs, and PNPP specific reviews. Table 3.1.1-2 of the submittal provides a complete list of the EPRI Transients, as listed in NUREG-2300. Table 3.1.1-5 provides classification of these transients into 5 Initiating Event Groups: T1, T2, T3A, T3B, and T3C. A total of three transients were discarded as they are not relevant to the PNPP and sufficient justification was provided in section 3.1.1. In addition to the IEs listed above, six other IEs were considered. These are: A, Large LOCA; S1, Intermediate LOCA; S2, Small LOCA; V, Interfacing Systems LOCA; O, Containment Bypass LOCA; and R, Vessel Rupture. This list of initiating events as well as the nomenclature is same as NUREG-4550 Grand Gulf Study, and is in general agreement with other BWR studies. The IE frequencies for these IPEs were essentially the same as for those used in the Grand Gulf Study. The only noticeable discrepancy is that IE frequency for T1 of 0.0609/yr. was lower than 0.11/yr. used in the Grand Gulf Study. This was attributed primarily to the differences in the off-site power distribution system. The only other difference is LOCA classification, which was discussed in section 3.1.1.1.2 of the submittal.

In addition to the eleven IEs listed above, two plant specific initiators were considered in the analysis. These are TIA, Loss of Instrument Air, and TSW, Loss of Service Water. These two initiators were retained after FMEA was reportedly used to eliminate failure of a variety of electrical systems, air systems, water systems and HVAC systems as initiators. The screening process used to obtain plant-specific initiating events is reasonable and consistent with PRA practices. However, it is not clear to the reviewer how PNPP IPE arrived at the IE frequencies used for these two plant specific initiates in the analysis. Specific discussions to this regard would be beneficial.

The IPE briefly described dependencies between initiating events and the mitigating functions and systems. The submittal did provide a complete dependency matrix for front line-to-support and support-to-support systems. Also, further details on the system dependencies were summarized in Section 3.2 of the IPE.

In summary, the initiating events selected were the same as those used in the Grand Gulf study, except for plant specific initiators. The IE frequencies were essentially the same as those used in the Grand Gulf study, except for plant specific initiators. In our judgment, the list of IEs, generic and plant-specific, is complete. Grouping of the IEs is consistent. The only possible deficiency is that the IPE does not clearly describe how IE frequencies for TIA and TSW were obtained.

II.1.2.2 Review of Front-Line and Support Systems Analysis

The list of frontline and support systems analyzed in detail are as follows.

Frontline Systems

RPV Depressurization
Standby Liquid Control
Residual Heat Removal
- Low pressure Coolant Injection Mode
- Containment Spray Mode
- Suppression Pool Cooling Mode
Low Pressure Core Spray
High Pressure Core Spray
Reactor Core Isolation Cooling
Condensate/Feedwater
Fire Protection Alternate Injection
ESW B/RHR B Cross-Tie Alternate Injection
Reactor Feed Booster Pump Alternate Injection
Containment Venting by Fuel Pool Cooling and Cleanup
Containment Venting by RHR Containment Spray

Support Systems

Suppression Pool Make-up

Drywell Vacuum Relief

ECCS Pump Room Cooling

Diesel Generator Room Ventilation

Emergency Closed Cooling

Nuclear Closed Cooling

Emergency Service Water

Safety Related Instrument Air

Service/Instrument Air

Emergency DC Power

Emergency AC Power

Service Water

Turbine Building Ventilation

Heater Bay Ventilation

Turbine Building Closed Cooling

For each of these systems, the IPE presented a brief system description, and details on system operation, system dependencies and interfaces, and system success criteria. Also provided are the schematics of each system.

From the review it is concluded that PNPP IPE analyzed all the important front-line and support systems required for prevention of core damage. It appears that the systems were modeled to the level of detail requested in NUREG-1335.

II.1.2.3 Systems Dependencies and Support Systems

Considerable effort was devoted in the PNPP IPE to identify important systems dependencies and interfaces (see Section 3.2.x of the IPE). Depending on the type of the system, related discussions covered areas such as power supply and control power, actuation, cooling water, and related operator actions. In addition, two dependency matrices were enclosed. The first one presented dependence of front-line systems on the support systems and the second one focused on support system-to-support system dependencies. The list of support systems analyzed was presented above. This includes the minimum required systems: electrical power, instrument air, HVAC, service water, and component water.

The IPE would have benefited from a description of the diesel supported fire water system the IPE took credit for fire water x-tie in several event trees.

Our review of the dependency matrix revealed several inconsistencies. *First, all the dependency matrix states that several of the frontline systems, including ADS and SRVs, are completely dependent on offsite electrical power.* This can not be true since these systems can be operated by the onsite D.G.s following loss of offsite power. This dependency should be changed to partial dependence. *The dependency matrix also indicates that a variety of frontline systems are partially dependent on the MCC switchgear HVAC.* The event tree for LOOP modeled the accident progression based on the assumption that MCC switchgear HVAC failure has little impact on the front line systems. If the event tree, which is based on more recent calculations is accurate, then the dependency matrix must be updated to reflect this new understanding.

From the review it is concluded that IPE treated dependencies between various plant systems in a consistent and reasonable manner. Few deficiencies were identified as discussed above. Specifically the dependency matrix should be updated to incorporate the comments presented above.

II.1.2.4 Treatment of Common Cause Failures

Description of techniques used to treat common cause failures is incomplete and/or missing. The IPE noted that detailed description is provided in Appendix C.2 which is missing from the submittal. The description could be improved and suggest close adherence to NUREG-4780 guidelines for common cause analysis. Additional description seems to indicate that Beta Factor method was used for common cause analysis.

II.1.2.5 Review of Event Trees

The PNPP IPE used functional event trees. A different FET was developed for: Loss of offsite power; transients with loss of PCS; transients with PCS initially available; transients with loss of feedwater, but with PCS initially available; inadvertent open relief valve on the RPV; loss of instrument air; loss of service water; large LOCA; intermediate LOCA; and small LOCA. The FETs were configured to model system response to specific initiating events through the use of event tree top logics.

The following paragraphs provide our specific comments related to each individual event trees.

<u>Transient with a Loss of PCS Event Tree</u>:
The event tree is essentially same as that developed in NUREG-4550. The event tree modeled all important steps of the accident mitigation. **No inconsistencies were found.**

<u>Transient with PCS Initially Available Event Tree</u>:
The event tree is essentially same as that developed in NUREG-4550. The event tree top U3 in this tree is different from U3 in the previous event tree. Confusion can be avoided if U3 in the previous tree

is named UB. Otherwise, the success criteria and event trees for both these transients are very similar. The event tree modeled all important steps of the accident mitigation. No inconsistencies were found.

Loss of Feedwater Transient Event Tree

The event tree is essentially same as that developed in NUREG-4550. The event tree modeled all important steps of the accident mitigation. No inconsistencies were found.

Large LOCA Event Tree:

All break sizes greater than or equal to 0.5 sq. ft. for a liquid break and greater than 0.3 sq. ft. for a steam break were classified as a Large LOCA Event Tree. This criteria is different from Grand Gulf study which modeled all breaks larger than 0.3 sq. ft. for both liquid and steam break. This deviation was adequately explained. The success criteria for LLOCA was based on MAAP code calculations. The event tree modeled the success criteria as well as the important accident mitigation events accurately. Only possible deficiency found was that the IPE may have not modeled closure of MSIV's and opening of SRV's were not modeled in the event tree. For a large LOCA outside the containment, failure to close the MSIV's will result in depletion of suppression pool water available for core. Our calculations indicate that substantial water depletion could occur in ten minutes after the LOCA. This includes water from the upper containment pool. It is possible that the IPE modeled this event. The IPE should specifically address this concern. No other deficiencies were found.

Intermediate LOCA Event Tree:

All break sizes between 0.01 and 0.5 sq. ft. for a liquid break and between 0.1 and 0.3 sq. ft for a steam break were classified as a Intermediate LOCA Event Tree. This criteria is also different from Grand Gulf study and the deviation was adequately explained. The success criteria for Intermediate LOCA was based on MAAP code calculations. The event tree modeled the success criteria as well as the important accident mitigation events accurately. Failure of MSIV's to close is important event for mitigating Intermediate LOCA. Water depletion rate is substantially larger than the make up rate. This concern should be addressed in the IPE. No other deficiencies were found.

Small LOCA Event Tree:

All break sizes less than or equal to 0.01 sq. ft for a liquid break and less than 0.1 sq. ft for a steam break were classified as a SLOCA Event Tree. This criteria is different from Grand Gulf study. This deviation was adequately explained. The success criteria for SLOCA was based on MAAP code calculations. The event tree modeled the success criteria as well as the important accident mitigation events accurately. In our judgment, failure of MSIV's to close is not important for mitigating SLOCA. No deficiencies were found.

In our judgment, these event trees adequately model all important mitigating actions to the level-of-detail required by the Generic Letter 88-20 and described in NUREG-1335. Deficiencies noted were discussed above.

Special Event Trees:

A total of five special event trees were developed for PNPP IPE. Our review comments on these event trees are listed below.

Loss of Offsite Power Event Tree:

The event tree top B1 should be "Onsite AC Power to Division 1 and Division 2" instead of "Offsite AC Power to Division 1 and Division 2". This is a simple misprint and should be corrected. Inclusion of an event tree top "cross-tieing of the Division 3 and Division 2 DGs" could be beneficial to model the accident progress. On the other hand Event tree top Hv models failure of MCC, Switchgear and Misc. Areas HVAC. The documentation cites recent calculations which indicate that Hv has little impact on the accident mitigation and thus a success probability of 1 was assigned for this event. Hence we recommend removing Hv from the event tree which would substantially simplify the accident progression. Otherwise the event tree has modeled all important mitigating actions. No other deficiencies were found.

Station Blackout:

Common cause failure of the DGs are the main contributors to station blackout. The event tree is essentially the same as that developed in NUREG/CR-4550. The event tree modeled all important steps of the accident mitigation. No inconsistencies were found.

Transient with Inadvertent Open Relief Valve:

The event tree is essentially the same as that developed in NUREG/CR-4550. The event tree modeled all important steps of the accident mitigation. No inconsistencies were found.

Loss of Instrument Air Event Tree:

In our judgment, the functional event tree is correct and no deficiencies were found.

Loss of Service Water Event Tree:

This event was not analyzed in NUREG-4550. In our judgment the functional event tree is correct and no deficiencies were found.

ATWS Event Tree

This is an important contributor to the total CDF. Important human action is failure to ADS Inhibit. A major difference between the Grand Gulf study and the PNPP IPE is the missing event tree top "Operator manually inserts individual groups of control rods". Otherwise the event tree is essentially same as that developed in NUREG/CR-4550. The event tree modeled all important steps of the accident mitigation. No inconsistencies were found.

Internal Flooding

See section II.1.1.4 for discussions on the methodology chosen and results of the Internal Flooding.

No event trees were developed for Interfacing System LOCA, Containment Bypass LOCA, and Vessel Rupture. Their contribution is expected to be less than 1.0E-8. In our judgment, the special events were treated rigorously, with the level-of-detail sufficient to reveal any vulnerabilities. We found no deficiencies.

II.1.2.6 Dominant Sequences

The point estimate for the CDF is 1.2E-5 /yr. The dominant IEs are: ATWS, LOOP, S.B., transients with the loss of PCS and loss of instrument air. Together these initiating events contributed about 95.4% of the total CDF. ATWS alone contributed about 41% of the total CDF. Although this is inconsistent with the NUREG-4550 results, it is acceptable given the following differences in the analyses.

1. Grand Gulf study took credit for operator manually inserting individual groups of control rods. PNPP IPE did not take credit for this operator action.
2. Grand Gulf study assumed that HPCS is an acceptable means of core injection. On the other hand Perry relied on BWR 6 owners group recommendation that HPCS should not be used to mitigate the accident.

These two deviations essentially contributed to the noted increase in ATWS contribution to CDF. The only other differences between the Grand Gulf study and the Perry IPE related to T2 and T1A. In both cases Grand Gulf CDF is lower than Perry values. But the deviations were adequately explained and appear to be correct.

The IPE identified and clearly discussed 15 dominant sequences that together contributed to about 81% of the CDF. Section 3.4.1.3 of the IPE present these discussions. Top ten dominant sequences that contribute to about 70% of the CDF are discussed below:

Sequence (T3A + T2)-C-U3-X'' (T2-cS30)

Frequency 2.27E-6 *Contribution 19.5%*

A transient has occurred. The PCS may be lost either due directly to the transient or due to subsequent conditions which may result in MSIV isolation. The control rods fail to insert into the core and the reactor remains at power. The motor feed pump has failed to inject into the RPV to maintain RPV level control. The Operators have failed to inhibit ADS resulting in rapid depressurization of RPV and injection of low pressure ECCS resulting in a reactivity excursion leading to core damage.

The initiating event T3A (transient with PCS available) contributes almost, three times as much to the failure of this sequence than does T2 (transient without PCS). This is due to the failure of the operators to maintain PCS available for an ATWS scenario. Following the initiating events T3A and T2 given the mechanical failure of the control rods, the dominant contributors to this sequence are failure of the operators to re-open the motor feed pump control valves, manually depressurize the RPV, and inhibit ADS.

Sequence T2-W-Y-Cv (T2S04)

Frequency 1.62 E-6 *Contribution 13.9%*

A loss of PCS transient has occurred with a reactor scram and successful SRV operation to maintain RPV pressure control. The motor feed pump has started and is successfully maintaining high pressure RPV level control. The RHR system and venting have failed to provide long-term containment heat removal. Without containment heat removal the containment ruptures disabling the injection path from the motor feed pump, thus leading to core damage.

The dominant contributors to the failure of this sequence are failure of the injection path upon failure of the containment and failure of 4,160 VAC Division 2 Bus EH12. The maintenance unavailability of RHR train A and the operator failure of the operators to align a containment vent path also contribute to the frequency of this sequence.

Sequence T1-B1-U1-Va-R (BS24)

Frequency 7.71 E-7 *Contribution 6.6%*

A loss of offsite power has occurred and the Division 1 and 2 diesel generators have failed to provide onsite AC power. HPCS has failed to provide high pressure RPV level control. RCIC has successfully provided high pressure RPV level control for 3 hours at which time the suppression temperature limit of 185° F has been exceeded and RCIC fails. The operators have successfully depressurized the RPV, but the fire protection system has failed to provide adequate low pressure RPV level control and offsite power was not recovered at 3 hours leading to core damage.

The dominant contributors to the failure of this sequence are unavailability of the fire protection system to provide alternate injection due to failure of the offsite power, failure of the diesel fire pump to run, and failure of the operators to align the fire protection system for RPV injection after RCIC is lost.

Sequence TIA-U1-U2-V-Va (TIAS14)

Frequency 7.53 E-7 *Contribution 6.5%*

A loss of instrument air has occurred with a reactor scram and successful SRV operation to maintain RPV pressure control. RCIC and HPCS have failed to provide adequate RPV level control at high pressure. The RPV has been successfully depressurized. With the RPV depressurized low pressure ECCS make-up and low pressure alternate injection have failed to provide RPV level control leading to core damage.

The dominant contributors to the failure of this sequence are failure of the operators to successfully align the reactor feed booster pumps or suppression pool cleanup for alternate low pressure injection. Common cause failure of Emergency Service Water pumps A and B, and other random failures of Emergency Service Water trains A, B, and C also contribute to the core damage frequency for this sequence.

Sequence (T3A + T2)-C-U3-X'' (T2-CS20)

Frequency 6.25 E-7 *Contribution 5.4%*

A transient has occurred. The PCS may be lost either due directly to the transient or due to subsequent conditions which may result in MSIV isolation. The control rods fail to insert into the core and the reactor remains at power. The motor feed pump has successfully injected into the RPV but the operators have failed to control RPV level. The operators have successfully inhibited ADS but have subsequently failed to initiate standby liquid control leading to core damage.

The initiating event T3A (transient with PCS available) contributes almost three times as much to the failure of this sequence than does T2 (transient without PCS). This is due to the failure of the operators to maintain PCS available for an ATWS scenario.

Sequence T-U1-R1-Ws-V-Va (RS20)

Frequency 6.04 E-7 *Contribution 5.2%*

A loss of offsite power has occurred with a reactor scram and successful SRV operation to maintain RPV pressure control. HPCS has failed, but RCIC has successfully provided high pressure RPV level control. At 3 hours RCIC failed due to failure of the Suppression Pool Cooling mode of RHR and non-recovery of offsite

power. The RPV has been successfully depressurized. Low pressure ECCS make-up and alternate low pressure make-up have failed leading to core damage.

The dominant contributors to the failure of this sequence are the failure of the operators to align fire protection for alternate injection after RCIC fails and the failure of the Division 3 diesel generator to start. Failure of the Division 1 and 2 diesel generators to start, the failure of the offsite power to provide adequate low pressure alternate injection, and the failure of the diesel driven fire pump also contribute to the failure of this sequence.

Sequence T1-B1-U1-U2-R-Va (BS34)

Frequency 5.26 E-7 *Contribution 4.5%*

A loss of offsite power has occurred, and the Division 1 and 2 diesel generators have failed to provide onsite AC power. HPCS and RCIC have failed to provide high pressure RPV level control. Offsite power was not recovered at 0.4 hours. The operators successfully depressurized the RPV, but fire protection alternate injection failed to provide adequate RPV level control.

The dominant contributors to the failure of this sequence are the failure of the offsite power, failure of the operators to bypass the RCIC isolation on high steam tunnel temperature, running failure of the diesel fire pump, failure of the operators to align fire water in a timely manner, and start failure of the division 3 diesel generator. Start failures of the Division 1 and 2 diesel generators also contribute to this sequence.

Sequence T1-B1-U1-R (BS17)

Frequency 3.36 E-7 *Contribution 2.9%*

A loss of offsite power has occurred, and the Division 1 and 2 diesel generators have failed to provide onsite AC power. HPCS has failed to provide high pressure RPV level control. RCIC has successfully provided high pressure RPV level control for 3 hours at which time the operators have successfully depressurized the RPV and aligned fire water as alternate low pressure injection. The batteries fail at 7 hours, and offsite power was not recovered by 13 hours. There is no containment heat removal leading to failure of the containment and subsequent failure of RPV injection leading to core damage.

The dominant contributors to the failure of this sequence are maintenance and starting failures of the Division 1, 2, and 3 diesel generators.

Sequence T1-U1-U2-R1-V-Va (US29)

Frequency 3.34 E-7 *Contribution 2.9%*

A loss of offsite power has occurred with a reactor scram and successful SRV operation to maintain RPV pressure control. HPCS and RCIC have failed to provide successful RPV level control at high pressure. Offsite power was not recovered by 0.4 hours, but the RPV has been successfully depressurized. Depressurization may be delayed until the MZIWL is reached dependent on the injection system alignment. With the RPV depressurized, low pressure ECCS make-up and fire protection alternate injection have failed to provide RPV level control leading to core damage.

The dominant contributors to the failure of this sequence are failure of the Division 3 diesel generator to start and maintenance of residual heat removal train A, LPCS, and RCIC. Failure of the offsite power and failure of the RCIC turbine driven pump also contribute to the core damage frequency for this sequence.

Sequence (T3A + T2)-C-U3-X (T2-CS28)

Frequency 3.12 E-7 Contribution 2.7%

A transient has occurred. The PCS may be lost either due directly to the transient or due to subsequent conditions which may result in MSIV isolation. The control rods fail to insert into the core and the reactor remains at power. The motor feed pump was not successfully placed into operation. ADS inhibit and standby liquid control are successful, but depressurization of the RPV by the operators was unsuccessful resulting in core damage.

The dominant contributor to the failure of this sequence is the failure of the operators to re-open the motor feed pump control valves and depressurize the RPV. For the IPE for a transient with PCS available coupled with an ATWS, it was assumed that the MSIV isolation at RPV level was not bypassed. This is also one of the dominant contributors to this sequence.

Based on our review, it is concluded that the IPE identified dominant sequences and expanded to the level of detail required to identify dominant contributors. In our judgment, the sequences are consistent with plant design.

II.1.2.7 Front-End and Back-End Interfaces

The Level 1/Level 2 interfacing was accomplished through a set of Plant Damage States (PDS). The PDS grouping logic diagram (Figure 3.1.4-19 of IPE) was used to group some PDSs together in order to reduce the number of required containment analyses. The grouping logic diagram asked a total of eleven questions listed below:

1. Not a containment bypass sequence
2. Containment status at core damage

3. Event Type

4. Initial Containment Heat Removal with Suppression Cooling

5. Containment Vent Isolated at RPV Failure

6. RPV Injection Failure Time

7. Offsite Power Recovery Time

8. Containment Heat Removal with RHR Spray Loop

9. Containment Heat Removal with Vent

10. Late In-vessel Injection and Pedestal Cavity Supply

11. RPV depressurized during core damage.

The logic diagram was checked for consistency. In our judgment, this grouping logic examined all the possible Level 1/Level 2 interfaces.

PDS Event trees were developed and submitted as part of the IPE for each event. These PDS event trees were developed by adding new event tree tops to the Level 1 event trees. PDS sequence quantification was performed using NUPRA. The PDS sequence screening criteria is truncation value of 1.0E-7/yr., as recommended in NUREG-1335. Based on the review the following conclusions have been drawn:

- Important sequences were not screened out. As noted above, a sequence was screened out only when its frequency fell below the truncation value of 1.0E-7 or sequence cut set frequency fell below 1.0E-10. This is consistent with NUREG-1335 guidelines.
- One of the top logics in the PDS grouping logic diagram (Figure 3.1.4-19) pertained to Containment by-pass.
- Plant Damage States explicitly considered all important reactor and containment systems. Additional top events were incorporated in the PDS event tree to include containment systems and some of the reactor systems and recovery actions into Level 1 Event trees.
- Source Term estimates could not be checked and could not be verified for consistency. This is primarily due to the fact that the IPE did not provide a clear definition of core damage states, i.e., percentage of core damage corresponding to each PDS. Given that, the only means of checking for the source term is to examine the MAAP output. This is out of our scope of work.
- System mission times, inventory depletion concerns and dual usage of sprays were accurately addressed. The only important issue is cavitation of HPCS pumps when the suppression pool reaches saturation. The IPE submittal clearly states that this is not a concern even when a maximum credible Large LOCA occurs, depleting the suppression pool.

- The analysis accurately modeled failure of containment heat removal leading to core injection failure. The containment fails before equipment failure occurs. This is similar to Grand Gulf and unlike Browns Ferry.

In our judgment, the Level I/Level II interfaces were accurately addressed and adequately documented. No deficiencies were found.

II.1.2.8 Multi-Unit Considerations

Perry Nuclear Power Plant consists of a single unit, Unit 1; the Unit 2 is incomplete and non-operational. There are no expected initiating events from Unit 2 that will effect Unit 1 operation. The only system shared between Units 1 and 2 is DC Batteries which are cross-tied. At present the cross-tieing is manual and efforts are underway to quicken this process. The IPE has taken credit for battery x-tie in the Station Blackout Event Tree. No other dependencies or cross-ties were found.

II.1.3 Review of IPE Quantitative Process

The PNPP IPE used PC based computer code NUPRA for sequence quantification and event tree / fault tree linking. The screening criteria listed in the Generic Letter 88-20 was used in the analysis. For example, a sequence cutset frequency truncation value of 1.0E-10 was used to screen-out some of the unimportant sequences. Recovery actions were included after the initial quantification by modifying the event trees. No sequences were cut off after adding the recovery actions. Additionally all the sequences with frequency higher than 1.0E-7 were retained for further analysis. Special attention was paid to analyzing IE that have important dependencies with the Front-line systems. The overall quantification process is widely accepted in the PRA community and no deficiencies were found. In addition, NUPRA was used for uncertainty analysis that quantified data uncertainty and its impact on CDF. Additionally, the uncertainty analysis focused on importance ranking of the basic events by Fussell-Vesely, Risk Reduction and Risk Achievement techniques. Finally, a sensitivity analysis was carried out to quantify the impact of initiating event frequency, human reliability, common cause failure data and maintenance data on the over all CDF. The results of the quantification process / uncertainty analyses are summarized below.

The point estimate for total CDF is 1.2E-5/yr. The distribution for CDF following uncertainty analysis is:

Mean:	1.4E-5/yr.
Standard Deviation:	3.9E-5/yr.
95th percentile:	2.5E-5/yr.
Median:	1.1E-5/yr.
5th percentile:	6.2E-6/yr.

The summary of sequences grouped by initiator are presented in the submittal as Table 3.4.1-2 and those contributing 95% of the CDF are presented as Table 3.4.1-3. The containment bypass sequences are given in Table 3.4.1-4.

In our judgment, the quantification process is sound and is based on methodology that is widely accepted in the PRA community. No deficiencies were found. A few inconsistencies in the data used are discussed below.

II.1.3.1. Quantification of the Impact of Integrated Systems and Component Failures

As mentioned previously, Fault Trees were developed for each of the front-line and support systems. Component failure data was obtained for each component. Fault trees were then integrated and linked to the event trees using NUPRA. However, a detailed uncertainty analysis was carried out to quantify the uncertainty in the data. Other comments on the data are presented below. In our judgment, the quantification process is good. No deficiencies were found.

II.1.3.2 Fault Tree Component Failure Data

The following sections discuss the submittal's treatment of fault tree component failure data. Separate discussions are presented concerning plant-specific, generic, and common cause component failure data. These discussions address Work requirements 1.3.2, 1.3.3 and 1.3.4 explicitly. The choice was made to integrate the discussions to minimize repetition.

Generic Data

The licensee has primarily used NUREG-4550 as the source for the generic failure data. We have spot checked failure data used for over 50 important components and found no noticeable inconsistencies. The PNPP IPE identifies the generic data as point estimates, where as the NUREG-4550 failure data are mean values. This inconsistency should be addressed or the discussion in the IPE should be changed to accurately identify the form of the data used.

It is concluded that the licensee has met the NRC's review guidance criteria regarding generic failure data used in the system fault trees.

Plant Specific Data

Guidance given on p. 2-6 (item 2.1.5.5) of NUREG-1335 states that plant-specific data generally should be used for certain types of items for plants with several years of experience unless a rationale is given. The

components recommended by NUREG-1335 include auxiliary feedwater pumps, emergency core cooling water pumps, batteries, electrical buses, breakers and diesel generators.

Because of the short operating history of Perry, the licensee has chosen to use generic data for most of the components listed above. However, the licensee stated that the diesel generators do have a failure history for some modes of failure and that a plant specific diesel failure rate is developed. It appears that the rate for running failures (7.86E-3) was derived from plant specific data. However, no method or model was presented in the submittal for the development of this value. Consequently we could not verify the accuracy of this value. Licensee should explain how they obtained data for failure to start in the IPE.

It is concluded that the use of generic data in the place of plant specific data is reasonable given the age of the plant. It was stated in the IPE that the licensee plans to update the IPE using plant specific data as it becomes available. However, no discussions were presented on how such a plan will be carried out.

The development of the running failure rate for the diesel generators was not justified. Considering that this value is nonconservative when compared to the NUREG/CR-4550 value (1.6E-02), it should be clearly justified.

Common Cause Data

The Perry IPE common-cause failure analysis was performed following the general guidelines of NUREG/CR-4780 (Mosleh, 1988). The Perry analysis relied on both a historical database (source: Haliburton NUS) and a quantitative screening process to obtain the dominant common cause component groups. The dominant common cause failure groups identified are listed below:

- ESW Motor Operated Valves
- Diesel Generators
- ESW Motor Driven Pumps
- ECC Motor Driven Pumps

The Perry analysis relied also on the Haliburton NUS failure database for data analysis. The generic common-cause failure events in the database were "re-integrated" to be Perry specific as recommended by NUREG/CR-4780.

The data analysis resulted in the following common-cause failure probabilities.

Component	Mean	—	EF	Median	95
ESW MOVs	9.234 E-5	1.82	20	1.762 E-5	3.524 E-4
ESW Pumps	3.413 E-4	0.81	3.6	2.455 E-4	9.340 E-4
ECC Pumps	1.291E-4	1.15	6.6	6.664 E-5	7.664 E-4
DGS	3.670 E-4	1.88	21	6.629 E-5	1.392 E-3

The methodology used for the Perry IPE common-cause failure analysis follows the procedures of NUREG/CR-4780 and is, therefore, acceptable for the IPE based on the guidance of NUREG-1335.

II.1.4 Review of IPE approach to reducing the CDF

II.1.4.1 Methodology for Identification of Plant Vulnerabilities

The PNPP IPE used NUMARC Severe Accident Issues Closure Guidelines for vulnerability screening. These guidelines are listed as follows:

> If the contribution from a given initiator or systems failure is greater than 50% to the total CDF, then it is interpreted as a significant vulnerability. If it contributes 20-50%, it is interpreted as a potential vulnerability to be investigated. Similarly, contribution from sequence groups between a core damage frequency of 1.0E-5 and 1.0E-4 are reviewed to determine if an effective plant procedure or hardware change which would reduce the frequency of the sequences.

Based on this vulnerability definition, the IPE concluded that no significant vulnerabilities exist. The IPE regrouped the sequences into a series of functional accident groups according to the criteria of NUMARC. Table 3.4.2-1 of the submittal presents these sequence groups and their contribution to CDF. Of these, two groups of sequences have been identified which contribute between 20-50%. The first group, referred to as Group 4 in the IPE, is made up of accident sequences involving ATWS leading to containment failure. The second group, referred to as Group 2 in the IPE, is made up of accident sequences involving loss of containment heat removal leading to containment failure and subsequently core injection failure. Section 3.4.2.1 presents discussions on these vulnerabilities and identifies specific systems/actions that contribute to these vulnerabilities. Several modifications have been proposed in response to these identified vulnerabilities, which are the matter of discussions in the following section.

In our judgment, the IPE has taken a reasonable approach to identifying plant vulnerabilities. No deficiencies were found.

II.1.4.2 Plant Improvements and Planned Modifications

The IPE listed following plant modifications that either have been completed or in the process of being implemented or to be implemented in the near future.

Modifications Already Implemented

Loss of Offsite Power Instructions

- Retention of RCIC isolation bypass for high steam tunnel temperature
- Enhanced process for cross-tieing Unit 1 and Unit 2 batteries
- Enhanced process for offsite power recovery to HPCS and alternate injection system buses

Flooding Instruction

- Enhanced response instruction for flooding scenarios

Modifications to be Completed in the Near Future

- ADS automatic initiation
- "Fast Firewater" tie between fire Protection and HPCS
- Permanent Division 3 to Division 2 "quick" connect
- Reduction of Out-of-Service Time for certain critical components

Modification to be Considered for Implementation

Passive Containment Vent Path

One of the sensitivity analyses performed was to assess the impact of containment failure on loss of RPV injection and subsequent core damage. The addition of a passive containment vent path that does not depend on AC power would reduce the core damage frequency from internal and flooding events by 18% or from 1.3E-5 to 1.1E-05

Automatic ADS Inhibit for ATWS

One of the contributors to core damage frequency for ATWS is manually inhibiting ADS. By installing an automatic inhibit of ADS, those ATWS sequences in which manual inhibit fails would drop out. The overall core damage frequency is reduced by 19% from 1.3E-5 to 1.0E-5. The sequences resulting from this failure result in an uncontrolled flow to the RPV from the low pressure injection systems with subsequent core damage and containment failure. The addition of the auto inhibit would reduce the frequency of this set of sequences.

In our judgment, the IPE has adapted a reasonable approach to identifying the plant vulnerabilities and planning appropriate plant modifications. The IPE has also expended adequate effort to quantify the impact of the modifications of the CDF.

II.1.5 Review of Licensee's Evaluation of DHR Function

II.1.5.1 IPE's Focus on Reliability of DHR

Based on the Vulnerability Screening, discussed above, and importance ranking measures examined for a potential vulnerability. As stated in the IPE, their contribution to CDF from loss of DHR is about 45% and comes primarily from the following functional failures:

Functional Failure	Percentage
Loss of Decay Heat Removal from Containment Leading to Core Injection Failure and Core Damage	22%
Loss of Offsite Power and Make-up	13%
Loss of Coolant Inventory Make-up at Low Pressure	8%
Loss of High Pressure Make-up and Failure to DEP	<1%

Based on the importance and sensitivity analyses the IPE concluded that a single plant modification that would significantly decrease this contribution is prevention of containment failure. The contribution of remaining individual components is small. A passive vent path is proposed in response and will be assessed in the future.

II.1.5.2 IPE Considered Diverse Means of DHR

IPE considered a diverse means of removing decay heat from the core including Feedwater Pumps, Motor Feedpump, RCIC, ADS&SRVs venting, HPCS, LPCS, and LPCI (once through and closed loops). It also considered reliance on the fire-water cross-tie as an alternative for low pressure injection. Similarly, the IPE considered Suppression Pool cooling, RHR Heat Exchangers, and containment venting as possible options for heat removal from the containment. **In our judgment, the IPE has considered all available diverse means of DHR. No deficiencies were found.**

II.1.5.3 Unique Features

1. Failure of containment leads to failure of core injection and Core Damage. Failure of RHR leads to failure of containment.
2. HPCS and RCIC switch-over from CST to Suppression Pool Cooling is automatic.
3. Fire water cross-tie is a standby core injection mode at low pressures.
4. Suppression Pool Makeup is by gravity head. The makeup rate can prevent NPSH failure of HPCS in the case of worst credible LOCA.

III OVERALL EVALUATION AND CONCLUSION

The PNPP IPE is a Level II PRA for all internal events and internal flooding only. The submittal includes brief system descriptions for both the front-line and support systems, except for the ESF Actuation System. Also given are line-drawn schematics of each system. The list of initiating events are complete and includes several "plant-specific" special initiators. The FMEA was used to screen out HVAC related initiating events. The initiator frequencies were reasonable and obtained from accepted sources (NUREGs and EPRI Reports). The fault trees were not enclosed to the submittal, which is not a requirement for "Step 1 Review". From the discussions it is clear that each system is modeled to the component and train level. The generic component failure data for a variety of components is taken from Grand Gulf study and appears to be reasonable and conservative. For several systems, including ECCS, generic data was used in place of plant specific data due to the short operating history. The common mode failures were reportedly handled according to the guidelines of NUREG/CR-4780. The methodology used appears to be adequate although we could not confirm it since the description of common cause analysis in the submittal is brief. Sequence quantification methodology and screening criteria are acceptable. An uncertainty analysis was carried out to quantify the impact of the data uncertainty on the CDF. The NUS-computer code NUPRA was used for SET/LFT linking, sequence quantification and uncertainty analysis. A listing of dominant sequences was provided and dominant contributors were explicitly discussed. An importance ranking of the functional events based on Fussell-Vesely, risk reduction and risk achievement techniques was provided. Explicit and detailed DHR analyses were presented. This analysis considered a diverse means of RHR removal from the core as well as from the containment. Finally, utility participation in the IPE as well as the utility peer-review process is reasonable and meets the intent of Generic Letter 88-20.

From the PRA it is clear that the CEI expended reasonable effort to gain insights and design plant modifications that will minimize plant vulnerabilities. Some of the plant modifications have already been completed and some are underway. Other plant modifications, which include "Automatic ADS Inhibit" and "Passive Containment Vent" are under consideration.

IV. IPE EVALUATIONS AND DATA SUMMARY SHEETS

This section includes the data sheets related to the front-end portion of the PNPP IPE. The format of this appendix follows that provided by the NRC in our task statement for Perry Nuclear Power Plant Unit 1. The section numbers are according to the NUREG-1335 Standard Table of Contents, which was closely adhered to by the IPE.

This appendix simply lists the data; no critique of the da[illegible] is presented here. This information is presented in the previous sections.

2.4 Information Assembly

Perry Unit 1 is a BWR/6 with Mark III containment. This unit is very similar to the Grand Gulf nuclear power station, which is a NUREG/CR-4550 reference plant. Other similar plants for which PRA studies have been performed include BWR/6s at Kuosheng in Taiwan and Cofrentes in Spain. All these plants have been cited in the IPE submittal. Very minor differences in the front-line systems exist between the Grand Gulf nuclear unit and Perry Unit 1. Most of the differences are related to cross-ties between units (DC batteries x-tie). But none of them were found to play significant role in the accident mitigation.

3.1.1 Initiating Events

Table 3.1.1-1 of the submittal provided the PNPP IPE initiating event list, which includes the mean frequency for each initiating event. Also, Table 3.4.1-1 of the IPE submittal provided a summary of core damage frequency by initiating event. These two tables were merged together to compile the following table.

3.1.2 Front-Line Event Tree Review

Licensee's basis for the Success Criteria

The licensee used MAAP analysis to develop success criteria for the Level 1 and the Level 2 PRA analyses. The success criteria are very similar to those listed in NUREG/CR-4550 for Grand Gulf, except they are slightly conservative in few cases. The success criteria are consistent with the information provided in the Updated Final Safety Analysis Report (USFAR).

Table IV.1. Contribution of Generic and Plant-Specific Initiators to the CDF

I.E.	Description	Freq/yr	CDF/yr	%
T1	Loss of Offsite Power Transient	0.0609	1.44E-6	12.4
B	Station Blackout	[a]	2.25E-6	19.3
T2	Transients with the Loss of the Power			
	Conversion	1.62	1.67E-6	14.3
T3A	System (PCS)	4.51	<1E-8	<0.01
T3B	Transients with PCS Initially Available			
	Transients Involving Loss of Feedwater;	0.76	<1E-8	<0.01
T3C	with			
	the PCS Initially Available	0.14	1.38E-7	1.2
TIA	Transients Caused by an Inadvertent Open			
	Relief Valve (IORV) on the RPV	0.092	1.01E-6	8.7
TSW	Transient Caused by a Loss of Instrument			
	Air	1.0E-3	6.7E-8	0.6
A	Transient Caused by a Loss of Service	1.0E-4	2.11E-7	1.8
S1	Water	3.0E-4	6.19E-8	0.5
S2	Large Loss of Coolant Accident (LOCA)	3.0E-3	3.34E-8	0.3
V	Intermediate LOCA	<1E-8	<1E-8	<0.01
O	Small LOCA	<1E-8	<1E-8	<0.01
R	Interface System LOCA	1.0E-7	<1E-8	<0.01
ATWS	Containment Bypass LOCA	[b]	4.74E-6	40.7
	Vessel Rupture			
	Anticipated Transient Without Scram			

Notes:

(a) Station Blackout is a separate event tree developed as part of LOOP (T1). It is not an I.E. by itself.

(b) All events leading to failure of scram following an I. E. are classified as ATWS. ATWS is not an I.E. by itself.

*I.Es in the bold are the plant specific I.E.s.

Functional vs Systemic Event Trees

The PNPP IPE used functional event trees. A different FET was developed for each I E listed in the table above. The FETs were configured to model system response to specific initiating events through the use of event tree top logics. Fault Trees were developed to model both front-line and support systems. Event tree top logics and fault trees are not a part of the IPE submittal.

HVAC Assumptions

The HVAC systems were reviewed for special initiators and were screened out through Failure Modes and Effects Analysis (FMEA). IPE cited engineering calculations which revealed that in spite of the failure of control room HVAC the control room temperature would remain less that 120° F, thus not challenging the control room equipment. Consequently no reactor trip is expected from the failure of the control room HVAC. Similar reasoning was provided for MCC, Switchgear and Misc. Electrical Equipment Area HVAC systems. Thus HVAC was screened out from the list of initiators.

Initial analyses, as reported (Page 3-25), apparently suggested that loss of MCC and Switchgear HVACs may result in breaker failure. Thus, this failure was formulated as a major event tree top event in the LOOP and Station Blackout event trees. Apparently more recent analyses revealed that this was not the case. Although this function has remained in the event tree, a success probability of 1 was assigned.

In summary, PNPP IPE assumed that none of the HVAC systems contribute to reactor trip and that none play a significant role in the accident mitigation. The IPE cited several utility calculations as the basis for these assumptions.

3.1.3 Special Event Tree Review

Perry Nuclear Power Plant is a BWR/6. RCP seal cooling is not relevant to this IPE.

3.1.4 Support System Event Tree

Event Tree Methodology

The PNPP IPE employed SET/LFT methodology.

Contractor Employed

The contractor employed by the PNPP is Halliburton NUS Corporation.

3.2.2 Fault Trees

Table 3.2.2-1 provides a detailed list of the front line and support systems for which fault trees were analyzed. It is reproduced below.

Frontline Systems

RPV Depressurization

Standby Liquid Control

Residual Heat Removal

- Low pressure Coolant Injection Mode
- Containment Spray Mode
- Suppression Pool Cooling Mode

Low Pressure Core Spray

High Pressure Core Spray

Reactor Core Isolation Cooling

Condensate/Feedwater

Fire Protection Alternate Injection

ESW B/RHR B Cross-Tie Alternate Injection

Reactor Feed Booster Pump Alternate Injection

Containment Venting by Fuel Pool Cooling and Cleanup

Containment Venting by RHR Containment Spray

Support Systems

Suppression Pool Make-up

Drywell Vacuum Relief

ECCS Pump Room Cooling

Diesel Generator Room Ventilation

Emergency Closed Cooling

Nuclear Closed Cooling

Emergency Service Water

Safety Related Instrument Air

Service/Instrument Air

Emergency DC Power

Emergency AC Power

Service Water

Turbine Building Ventilation

Heater Bay Ventilation

Turbine Building Closed Cooling

3.2.3 System Dependencies

Plant Unique Dependencies

The dependency matrix indicates that several of the front-line systems are completely dependent on offsite power. These include ADS and SRV, among numerous others. This can not be true since these systems receive emergency power, and the dependency matrix must be corrected. Similarly, the dependency matrix indicates partial or delayed dependence of several front line systems on the HVAC systems. The IPE on the other hand states that recent calculations have shown this to be untrue. In which case the dependency matrix must be corrected. Otherwise, the only unique dependencies noted in the review are as follows:

1. The safety-related dc buses can be cross-tied to the dc batteries in unit 2, which is incomplete. This feature extends availability of dc power following LOPA and Station Blackout sequences.
2. The HPCS D.G. can be cross-tied to Division 2 emergency bus, which enables the containment vent valves and hydrogen ignitors to be powered in the event of LOOP and loss of Div. 1 and 2 D.G.s.
3. Core injection as a function is dependent (partial and delayed) on the containment heat removal systems. This is not adequately represented in the IPE.
4. None of the front-line systems are dependent on the HVAC systems.

Plant Asymmetries

The Division 3 HPCS Diesel Generator is a different size and type from the Divisions 1 and 2 DGs. Any other asymmetries present are those implicit to BWR/6 design.

3.3.1 List of Generic Data

NUREG/CR-4550 was adopted as the primary source for the generic data. Only slight differences were noted between the generic data for Initiating Frequencies reported in NUREG/CR-4550, which were taken from NUREG/CR-3862, and the PNPP IPE. The generic failure data for various components is checked against other sources, including NUS BWR Generic Data, NUREG/CR-1363, NUREG/CR-1740, NUREG/CR-3831, IEEE-500, WASH-1400, and GESSAR II. Our review revealed little difference between the Generic Data used in the PNPP IPE and NUREG/CR-4550 for most components.

3.3.2 Plant Specific Data and Analysis

Plant specific data was used for failure rate of the diesel generators. Also system unavailabilities from testing and maintenance of ECCS and RCIC were derived based on plant specific data. However, due to lack of plant specific data for failure of other systems, such as ECCS pumps, generic data was used. The IPE submittal did not provide specific sources of this generic data. Additionally, IPE did not provide methodology (e.g. Bayesian

Updates) used to derive the plant specific data for the cases where it was derived from plant specific data. Such information must be provided as part of the IPE for review. In some cases the plant specific failure data appears to be grossly inconsistent with the NUREG/CR-4550 data. It should be better explained.

3.3.4 Common Cause Failure Analysis

Techniques Used to Treat Common Cause Failures

Description is incomplete and/or missing. The IPE noted that detailed description is provided in Appendix C.2 which is missing from the submittal. The description provided is inadequate and suggest close adherence to NUREG-4780 guidelines for common cause analysis. Additional description seem to indicate that Beta Factor method was used for common cause analysis.

Level of Treatment

One again this information is not provided in the IPE. It is assumed that IPE followed NUREG-4780, which would indicate that the level of treatment is component groups.

Most Significant Common Cause Failures

The following were listed in the IPE as the most significant common cause failures.

System	Perry Value
ESW MOVs	9.234E-5
Diesel Generators	3.67E-4
ESW Motor Pumps	3.416E-4
ECC Motor Pumps	1.291E-4

Sources of Common Cause Data

Sources of the data were not explicitly stated in the IPE. Our review has clearly demonstrated that failure data is significantly different from the NUREG/CR-4550 common cause failure data. One of our recommendations is to provide sources of this data.

3.3.5 Quantification of Unavailability of Systems and Functions

Systems or Components with Noted Unusually High or Low Unavailability

No systems or function were found to have unusually large or low unavailabilities.

Sources of the Data

Once again, the sources of the data are not clearly identified. In Section 3.3.2 it was mentioned that Perry Unit 1 documents were used to determine plant specific unavailability data for ECC and RCIC systems. However, methodology used was not specified. Sources of the data for remaining components was not listed. It appears that at least for some components NUREG/CR-4550 data was used. This remains to be confirmed by the utility.

3.3.7 Quantification of Sequence Frequencies

Codes Employed

The entire Level 1 analysis including the internal flooding was performed on the NUPRA workstation, developed and supported by Halliburton NUS Corporation.

Uncertainty Analysis

Scope. An uncertainty analysis was performed to evaluate the uncertainty on CDF resulting from the uncertainties on the parameter values of the core damage model. In addition, a sensitivity analysis was conducted to quantify the impact of uncertainty in initiating event frequency, success criteria, human reliability, common cause failure data and maintenance data on the overall CDF.

Method. It appears that NUPRA was also used for uncertainty propagation.

3.3.8 Internal Flooding

Methodology

A preliminary screening analysis followed by a more in-depth analysis on those areas not screened was employed. The screening analysis is similar to the IDCOR method where the plant buildings are broken up into flood zones and vital safety equipment is identified within each zone. Major flooding sources are then identified within the zones and flooding initiator frequencies are calculated from generic data from U.S. nuclear plant experience. For the screening analysis, vital safety equipment located within the flood zone is assigned a failure probability of 1 and a conditional failure probability of the remaining safety equipment required to lead to core damage. The frequency of core damage appears to have been screened against a frequency of 3 X 10-7/yr, with the surviving scenarios then being more thoroughly analyzed.

Contribution to CDF

A total contribution of 1.5 X 10-6/yr, representing 12% of the total contribution from internal events and flooding was determined.

Critical Internal Flood Areas:

From Table 3.3.7-1, nine areas survived initial screening. They consist of:

1. Control Complex - 576' 10"
2. Control Complex-599',
3. Turbine Building and Turbine Power Complex,
4. Auxiliary building-568',
5. Auxiliary Building-599',
6. Steam Tunnel,
7. Control Complex, Unit 1 Division1/2, Battery Room/Switch gear -638.5"
8. Auxiliary Building Corridors,
9. Service Water Pumphouse

Only five of these areas are subsequently discussed as having been analyzed in detail. Of the nine surviving areas, only the first four appear to be significant contributors to core damage accounting for a total of 98% of the internal flooding contribution to core damage.

Most Critical Flood Sources:

Service Water, Circulating Water (Turbine Building), Emergency Service Water, Condensate Transfer

3.4.1 Screening Criteria

Screening criteria listed in the Generic letter 88-20 was used in the analysis.

Form of Truncation

A truncation value of 1.0E-10 for the sequence cutsets was used. In addition, every sequence with a probability of 1.0E-7 were retained for further analysis.

Definition of Core Damage

Core damage was defined as failure to maintain the water level in the RPV above the Minimum Zero Injection Water Level or, in the case of ATWS, failure to maintain the maximum cladding temperatures below 2200 F, with no possibility of recovery of injection in the short term.

Total Core Damage Frequency

The point estimate for total CDF is 1.2E-5/yr. The distribution for CDF following uncertainty analysis is:

Mean: 1.4E-5/yr

Standard Deviation:3.9E-5/yr

95th percentile:2.5E-5/yr

Median:1.1E-5/yr

5th percentile:6.2E-6/yr

Dominant Contributors

The dominant contributors to the risk are: ATWS, Transients (mostly T2 and TIA), LOOP, and Station Blackout. Together they contribute about 97.4% of the total CDF.

Recovery Actions

The following human recovery actions were considered in the analysis:

1. Operator fails to X-tie Unit 1 and Unit 2 batteries
2. Operator fails to align condensate transfer alternate injection
3. Operator fails to align fire protection after RCIC fails due to suppression pool temperature
4. Operator fails to align fire protection after RHR fails due to MCC HVAC failure
5. Operator fails to align fire protection after HPCS fails
6. Operator fails to align fast fire protection alternate injection
7. Operator fails to control reactor feed booster pump during a loss of instrument air transient
8. Operator fails to control the reactor feed booster pump following loss of Instrument Air in a time frame greater than 2 hrs
9. Operator fails to align suppression pool clean up alternate injection (late injection)
10. Operator fails to locally open 1G41-F0145
11. Operator fails to depressurize after core damage having failed to depressurize early
12. Operator fails to initiate SLC given early failure to initiate.

Only other recovery action is recovery of offsite power in the case of LOOP.

3.4.2 Vulnerability Screening

Importance or Relative Ranking Provided?

Yes. Initiating Event Importance Ranking by Fussell-Vesely, Risk Reduction and Risk Achievement methods were provided in the IPE (Tables 3.4.1-7).

Licensee's Definition of Vulnerability

If the contribution from a given initiator or system failure is greater than 50% of the total CDF it is interpreted as a significant vulnerability. If it contributes between 20 and 50% it is interpreted as a potential vulnerability to be investigated. Additionally, contribution from sequence groups between a CDF of 1.0E-5 and 1.0E-4 are reviewed to determine if there is an effective plant procedure or hardware change which would reduce the frequency of the sequences.

Vulnerabilities

No initiators or a group of sequences resulted in CDF larger than 50% of the total. Hence no significant vulnerabilities were found. However, ATWS sequences (Group 4) and loss of containment heat removal sequences (Group 2) contribute between 20-50% of the total CDF. In the ATWS sequences the potential vulnerability is the failure to inhibit ADS, which leads to reactivity oscillations and possible core damage. In the Group 2 sequences the potential vulnerability is due to the fact that loss of containment heat removal also results in loss of core injection. The contribution of these sequences is about 2.6E-6, which is well below 1.0E-5. None of the vulnerabilities are RHR related or Internal Flooding related.

Plant Fixes

Two potential plant improvements were considered to reduce the core damage frequency. These are:

1. Passive Containment Vent Path which reduces CDF from 1.3E-5 to 1.1E-5 (19% reduction).
2. Automatic ADS Inhibit for ATWS which reduces CDF from 1.3E-5 to 1.0E-5.

However, these proposed improvements are under consideration and their necessity will be determined following future updates to the IPE. No schedules were proposed for this change.

Plant Life Extension

IPE did not consider plant life extension in the proposed plant modifications.

3.4.3 Decay Heat Removal

Method of DHR

IPE considered a diverse means of removing decay heat from the core including feedwater pumps, motor feedpump, RCIC, ADS&SRVs venting, HPCS, LPCS, and LPCI (once through and closed loops). It also considered reliance on the fire water cross-tie as an alternative for low pressure injection. Similarly, the IPE considered suppression pool cooling, RHR heat exchangers, and containment venting as possible options for heat removal from the containment. Thus, the IPE has considered all available diverse means of DHR.

Credit for Recovery of PCS

IPE does not take credit for recovery of power conversion systems

Main Feed Water Trip on Reactor Trip

No Credit was taken in some sequences for continued operation of the main feed water pumps after reactor trip.

Unique Front-end System Features

Important unique features include:

1. There is a motor driven feed pump that is normally in standby and will start on an automatic signal at Level 2, following failure of the turbine driven pumps.
2. The safety-related dc buses can be cross-tied to the dc batteries in unit 2, which is incomplete. This feature extends availability of dc power following LOPA and Station Blackout sequences.
3. The HPCS D.G. is not of same size or type as the other two D.G.s. This will reduce likelihood of common cause failure of all three D.G.s.
4. The HPCS D.G. can be cross-tied to Division 2 emergency bus, which enables the containment vent valves and hydrogen ignitors to be powered in the event of LOOP and loss of Div. 1 and 2 D.G.s.
5. Containment failure leads to injection failure.
6. Makeup to the Suppression Pool is provided by gravity head. No pumps are involved.
7. HPCS and LPCS inject inside the shroud. Consequently, the HPCS is not a recommended system for ATWS mitigation.
8. The "ADS inhibit" action is not automatic.

6. Plant Improvements and Unique Safety Features

Important Insights

The IPE results indicate that ATWS is the largest contributor to the CDF. This result is different from Grand Gulf study, where the major contribution was by common cause failure of the DGs. Major reasons for the deviation are:

1. PNPP IPE did not take credit for operator manually inserting individual groups of control rods, and
2. PNPP IPE assumed that HPCS injection is no longer an acceptable means of injecting water into the core following ATWS.

The largest contributor to ATWS sequences is from operator failure to inhibit ADS. This operator failure would lead to reactivity oscillations and subsequently core damage.

The second largest contributor to the CDF is failure of core injection upon failure of containment heat removal. The third largest contributor is common cause failure of the Division 1 and 2 DGs. This latter one is further complicated by the fact that Division 3 DG cross-tie to Division 2 is very limited in comparison to Grand Gulf. Proposed plant improvements in response to these insights are listed below.

Implemented Plant Improvements or Enhancements

The following plant improvements were made as a result of the IPE:

1. Retention of RCIC isolation bypass for high steam tunnel temperature
2. Enhanced process for cross-tieing Unit 1 and Unit 2 batteries
3. Enhanced process for offsite power recovery to HPCS and alternate injection system buses
4. Enhanced response instructions to flooding scenario
5. Reduction of Out-of-Service Time for certain critical components

Plant Improvements for Which Credit Has Been Taken

None

Plant Improvements Under Consideration

The following plant improvements are proceeding although exact schedule for completion are not provided. These are:

1. Fast Firewater tie between Fire Protection and HPCS
2. Permanent Division 3 to Division 2 "quick" cross-tie

Additionally, the following plant improvements are under consideration:

1. Passive Containment Vent Path
2. Automatic ADS Inhibit for ATWS.
3. Systematic Maintenance Optimization.