ENCLOSURE 3

PERRY INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(BACK-END)

ERI NRC 92-1106

STEP 1 TECHNICAL EVALUATION REPORT OF THE PERRY INDIVIDUAL PLANT EXAMINATION (IPE) BACK-END SUBMITTAL

FINAL

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ACKNOWLEDGMENT

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REFERENCES

1. USNRC, "PRA Procedures Guide," NUREG/CR-2300, January 1983.

2. Swain, Alan D., "Accident Sequence Evaluation Program Human Reliability Analysis Procedure," NUREG/CR-4772, February 1987.

3. Swain and Guttman, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," NUREG/CR-1278, August 1983.

ENCLOSURE 5

SUMMARY OF THE PERRY NUCLEAR POWER PLANT

INDIVIDUAL PLANT EXAMINATION (IPE)

SUBMITTAL ON INTERNAL EVENTS

Summary of the Perry Nuclear Power Plant Individual Plant Examination (IPE) Submittal on Internal Events and Internal Flooding.

The NRC staff completed its review of the internal events portion of the Perry nuclear power plant Individual Plant Examination (IPE) submittal, and associated documentation which includes licensee responses to staff generated questions and comments. The licensee's IPE is based on a Level 2 PRA consistent with guidance issued in Generic Letter 88-20 Appendix 1. The IPE was performed by a team of personnel from Cleveland Electric Illuminating Company (CEI), Halliburton NUS Environmental Corporation, Gilbert Commonwealth, and Garbor Kenton and Associates. CEI maintained involvement in the development and application of PRA techniques to the Perry facility with the objective of transfer of PRA technology to the CEI licensee personnel. The Perry IPE represents the currently as-built and as-operated plant. The staff notes that major plant departments provided input to the IPE development as part of a peer review. In addition, the licensee intends to maintain a "living" PRA.

Perry has several features that impact the core damage frequency (CDF). Features that have a positive impact on CDF include: a stand-by motor driven feed pump, dc bus cross-tie capability with the (uncompleted) Unit 2 dc batteries, different size HPCS DG than the other two DGs (diversity reduces DG common cause failures); cross-tie capability of the HPCS DG to the Division 2 emergency bus. Features that have a negative impact include: containment failure leads to injection failure, did not take credit for HPCS injection as part of the success criteria for ATWS, and "ADS inhibit" action is not automatic.

The licensee used NUMARC Severe Accident Issues Closure Guidelines to identify vulnerabilities. According to these guidelines if the contribution from a given initiator or systems failure is greater than 50% of the total CDF, then it is interpreted as a significant vulnerability. If the contribution is 20-50%, it is interpreted as a potential vulnerability to be investigated. Similarly, sequences with frequencies between 1.0E-5 and 1.0E-4 were reviewed by the licensee to determine if an effective plant procedure or hardware change would reduce these frequencies. No vulnerabilities were identified based on this definition.

The Perry IPE results are different from the Grand Gulf NUREG-1150 study, where the major contributor was common cause failure of the DGs. The differences stem primarily from the fact that the Perry IPE (a) did not take credit for manual insertion of individual groups of control rods and (b) did not assume HPCS injection as part of the success criteria for ATWS.

Based on the review of the Perry IPE submittal and associated documentation, the staff finds the licensee's conclusion that no fundamental weakness or severe accident vulnerabilities exist at Perry acceptable and concludes that the licensee met the intent of Generic Letter 88-20.

The licensee's IPE results are (*) summarized below:

- o Plant type: BWR 6
- o Containment Type: Mark III
- o Total Core Damage Frequency (internal events and flooding): 1.32E-5/year
- o Flooding alone: 1.54E-6
- o Major Initiating Events and contribution to core damage frequency (internal events):

	Contribution
LOSS OF OFFSITE POWER	12.4%
LOCAs	2.6%
BLACKOUT	19.3%
ATWS	40.7%
TRANSIENTS	25%

- o Major contributions to dominant core damage sequences:
  - Transient followed by failures of PCS, the control rods, the feedwater pump, and the operator to inhibit ADS (19.5% of the total CDF).
  - Loss of PCS followed by failure of containment heat removal through RHR and containment venting (13.9%).
  - Station blackout followed by failure of HPCS, recovery of offsite power within 3 hours, the fire protection system (6.6%),
  - Loss of instrument air followed by failures of RCIC and HPCS and of the low pressure ECCS and low pressure alternate injection systems (6.5%),
  - Transient, followed by failure PCS, the control rods, and the operator to initiate the standby liquid control (5.4%),
  - Loss of offsite power followed by failures of HPCS, suppression pool cooling, recovery of offsite power, and of the low pressure ECCs and alternate low pressure make-up (5.2%).
  - Station blackout followed by HPCS and RCIC failures, non-recovery of offsite power in 0.4 hours and the fire protection system to provide alternate injection. (4.5%)
  - Station Blackout followed by failures of HPCS, the batteries at 7 hours and recovery of offsite power within 13 hours, resulting in failure of containment heat removal and coolant injection. (2.9%)
  - Loss of offsite power followed by HPCS and RCIC failures, non-recovery of offsite power in 0.4 hours and failure of the low pressure ECCS make-up and the fire protection system to provide alternate injection. (2.9%)
  - Transient, followed by failure of the PCS, the control rods, the operator to re-open the motor feed pump and depressurize the reactor. (2.7%),

- Transient, followed by failure of the PCS, the control rods, and the operator to initiate the standby liquid control (2.5%),
- Transient, followed by failure of the PCS, the control rods, and the operator to inhibit ADS, re-open the motor feed pump control valves and depressurize (2.4%).
- Loss of instrument air followed by failure of RCIC, and containment heat removal through RHR or containment venting (2.2%).
- Transient, followed by failure of the PCS and the control rods and the operators fail to inhibit ADS (2.0%).
- Large LOCA followed by HPCS and low pressure ECCS. Contributors are failure of the or unavailability of the Emergency Service Water.

o Significant PRA findings:

  - Loss of injection (both high or low pressure) (68%),
  - Failure of Containment Heat Removal (28%),
  - Failure to Recover Offsite Power (22%),
  - Containment Failure (22%), and
  - Failure to inhibit ADS following ATWS (25%).

Note: The above percentages indicate the % of the sequences that involve the function under consideration and not the absolute % contribution of the specific function to CDF. Therefore above percentages are not additive.

o Major operator action failures:

Failure to:

- Maintain the Power Conversion System (PCS) during ATWS,
- Re-open the motor feed pump control valves and manually depressurize the reactor during ATWS,
- Inhibit ADS during ATWS,
- Align a containment vent path,
- Align the fire protection system,
- Align the reactor feed booster pumps or suppression pool cleanup for alternate low pressure injection,
- Initiate standby liquid control, and
- Bypass the RCIC isolation on high steam tunnel temperature.

o Plant modifications implemented or under way:
- Loss of offsite power procedure improvement for retention of RCIC isolation bypass for high steam tunnel temperature,
- Loss of offsite power procedure improvement for cross-tying Unit 1 and Unit 2 batteries, and recovering offsite power to HPCS and alternate injection system buses,
- Procedural improvements for responding to flooding scenarios,

- Maintenance improvements for reducing out-of-service time for certain critical components.
- "Fast firewater" tie between Fire Protection and HPCS for alternate emergency cooling injection, and
- Permanent Division 3 to Division 2 "quick" crosstie.

o Modifications under evaluation:

  - ADS automatic initiation (other than ATWS),
  - Passive Containment Vent,
  - ATWS/ADS automatic inhibit,
  - ATWS/Feedwater runback between MSCWL and the Level 2 MSIV isolation bypass, and
  - Alternate Boron Injection.

* Information has been taken from the Perry IPE and has not been validated by the NRC staff.

1. INTRODUCTION

This Technical Evaluation Report (TER) documents the results of a Step 1 review of the Perry Individual Plant Examination (IPE) Back-End submittal [1], based on the Step 1 review objectives set forth by the NRC. These objectives include the following:

- To determine if the IPE submittal essentially provides the level of detail requested in the Submittal Guidance Document, NUREG-1335
- To assess the strengths and the weaknesses of the IPE submittal
- To provide a preliminary list of questions based on this limited review
- To complete the IPE Evaluation Data Summary Sheet.

This TER complies with the requirements of the contractor task order for Step 1 review. Section 2 summarizes our findings and briefly describes the submittal as it pertains to the work requirement. Each portion of Section 2.1 corresponds to a specific work requirement as outlined in the NRC contractor task order. Identified IPE submittal strengths and weaknesses are summarized in Section 2.2. Section 2.3 contains a review of the principal issues addressed in the licensee response to NRC questions, which were forwarded to the licensee following the issuance of the draft version of this report. Section 3 contains a summary of the overall IPE evaluation and the review conclusions. The Appendix contains the IPE evaluation summary sheets.

2. CONTRACTOR REVIEW FINDINGS

The present Step 1 review compared the Perry IPE submittal to the requirements of Generic Letter 88-20 (GL), according to the guidance provided in NUREG-1335. The findings of the present Step 1 review are reported in this section. The review findings, based on NRC's acceptance criteria for IPE submittals, are divided into submittal strengths, submittal weaknesses, and issues that can only be resolved through further interactions with CEI. In Section 2.3 of this report, questions are posed for NRC staff use in further review of the IPE.

The key points of the GL (and its Supplements) have been addressed in the IPE submittal report. However, some areas of concern do exist. These are discussed in Section 2.2.

2.1 Review and Identification of IPE Insights

2.1.1 General Review of IPE Back-End Analytical Process

2.1.1.1 Completeness

The IPE submittal is essentially complete with regard to the requirements of the Generic Letter and NUREG-1335.

2.1.1.2 Description, Justification and Consistency

The IPE methodology used for the back-end evaluation follows the procedure of the NUREG-1150 analysis of the Grand Gulf plant [2]. The entire process, including the definition of the plant damage states, the Containment Event Tree (CET) analysis, and the definition of source term bins, represents a condensed version of the NUREG-1150 analysis. It is important to note that the Perry IPE back-end analysis is much more transparent and easily understandable than the NUREG-1150 analysis, with no apparent loss of completeness. The approach followed is consistent with generic letter GL 88-20.

2.1.1.3 Peer Review of the IPE

Section 5.2 provides a discussion of the independent review of the IPE submittal. No specific issues related to the back-end analysis were identified by the reviewers.

2.1.2 Containment Analysis and Characterization

The Perry plant has a General Electric BWR/6 reactor rated at 3579 MWt and a Mark III steel containment. A description of the containment is provided in Section 4.1.1 and Appendix H.1 of the submittal, and Figure 4.1.1-1 illustrates some of the design features of the containment.

The Perry containment vessel is a free standing steel shell with an elliptic dome located inside a concrete shield building. The drywell is inside the containment vessel. All of the structures are supported by a concrete mat. The steel containment vessel is designed for a maximum internal design pressure of 15 psig with a coincident temperature of 185°F, and a maximum external pressure of 0.8 psid. The drywell is a vertical cylinder connected to the suppression pool by 120 vents. Table 4.1.1-1 in the submittal provides a summary comparison of the key design features of the Perry and Grand Gulf plant and containment systems.

2.1.2.1 Front End/Back End Dependencies

The interface between the front-end and back-end analyses is defined by a set of Plant Damage States (PDSs). The plant damage states define a set of functional characteristics for system operation which are important to accident progression, containment failure and source term definition. Each plant damage state defines a unique set of conditions regarding the state of the plant and containment systems, the state of the core and the Reactor Coolant System (RCS), and the state of the containment boundary at the time of core damage.

The Perry IPE plant damage states are defined based on the following combination of binning characteristics:

1. Initiating Event Type Identifies the accident initiator. Types of accident initiators include Station Blackout (SBO), loss of offsite power, Anticipated Transient Without Scram (ATWS), and all other transients and LOCAs, which are classified under the sub-heading of 'other' accident sequences.

2. Containment Status at Core Damage Identifies the containment status at core damage as either intact, isolated, bypassed or failed.

3. Timing of Power Recovery for SBO Sequences Indicates the timing of offsite power recovery for station blackout sequences (i.e., prior to vessel failure, prior to containment failure, or no recovery).

4. Status of Containment Mitigation Features Differentiates accident sequences based on the functional availability of containment mitigation features such as containment sprays, containment heat removal, and containment venting.

5. Reactor Pressure Vessel Injection Status Indicates the reactor pressure vessel (RPV) injection failure time. Also, indicates the possibility for late in-vessel injection (prior to core plate failure).

6. Pressure of the Reactor Pressure Vessel Indicates whether or not the RPV is depressurized during core damage.

The assignment of the end states of the Level 1 event trees to the corresponding plant damage states was performed by extending those event trees through a plant damage state grouping logical diagram (displayed in Figure 4.3.2-1 of the submittal). The plant damage state event tree has 11 top events including the following: status of the containment, availability of containment heat removal, containment heat removal with vent, late RPV depressurization, and late in-vessel injection and the recovery of AC power. It was found that 75 possible plant damage states were required to group all the Level 1 accident sequence outcomes. Plant damage states with frequencies less than 10^{-8} were eliminated from further back-end analysis. A listing of the 15 plant damage states contributing 95.2% of the total core damage frequency (including both internal and flooding sequences) is provided in Table 4.3.3-1 of the submittal. The PDS binning process appears to be reasonable, and includes all indicators of interest to back-end analysis.

2.1.2.2 Sequences With Significant Probability

Probabilistic quantification of severe accident progression for plant damage states is performed using a condensed version of the Grand Gulf containment event tree (CET) developed as a part of NUREG-1150 [2]. The actual event tree is too detailed to be listed here, and is provided in Table 4.5.1-1 of the submittal. Severe accident progression was classified into the following four general time periods of interest: (1) initial, which involves the time period prior to core damage (for entry into the tree), (2) early, which involves the time period from the beginning of core damage to just before vessel breach, (3) intermediate, which involves the time period from immediately before vessel breach to the time of significant core concrete interaction, and (4) late, which involves the time period during the time of significant core concrete interaction. A total of 125 questions covered the accident progression in the Grand Gulf CET, whereas the event tree in the Perry IPE submittal has 68 nodal questions. A summary containment event tree is presented in Fig. 4.5.2.1-1 of the submittal, but this tree groups together several of the nodes in the actual event tree, and is for the purpose of understanding only. Descriptions of each event and the rationale for assigning particular split fractions to CET nodes are provided in Appendix H (Section H.3).

The first 11 events in the Perry CET cover the definition of plant damage states, and have already been addressed in Section 2.1.2.1. It is to be noted that although the corresponding portion of the Grand Gulf CET has 22 questions, all of the important issues are addressed by the CET found in the submittal.

The next 18 events analyze the early time frame, from the time of core damage to the time just prior to vessel breach. Important phenomena treated include coolability of debris in-vessel, containment failure before vessel breach (principally due to hydrogen combustion), and injection and spray failure due to containment failure. Some important phenomena not treated include the probability of the Steam Relief Valves (SRVs) sticking open (during cycling), and the probability of the core reaching a critical configuration after core damage. The phenomena of recriticality of the core after core damage is important for BWRs, especially for ATWS sequences. A probability of 0.1 was assigned in NUREG-1150 for the core reaching a critical configuration after injection.

Twenty-four nodes in the event tree address the intermediate time frame, from just prior to vessel breach until the time prior to the beginning of core-concrete interaction. Potential modes of drywell or containment failure considered include α-mode steam explosions, pedestal cavity steam explosions, high pressure melt ejection induced direct containment heating, hydrogen combustion, and steaming. In addition, suppression pool bypass is treated by two event tree nodes. Four nodes address spray and injection failure due to containment failure. One important point to be noted here is that the probability used in the submittal for an α-mode explosion-induced bottom-head failure was taken from NUREG-1150, but was arbitrarily reduced by a factor of ten. The submittal did check the impact of this parameter by performing a sensitivity analysis, the results of which showed a 58% decrease in No RPV Failure, a 10% decrease in No Pool Bypass, and a 1% increase in Containment Failure.

The last 15 nodes in the tree address events that occur in the late phase, i.e., during the core-concrete interaction phase of the accident. Core concrete interactions in the pedestal cavity, the associated pedestal structural failure, and the generation and possible combustion of hydrogen are all considered in the event tree. However, the treatment of combustion of CO in the event tree is not clear. Late suppression pool bypass is also treated. Late venting of the containment and the possibility of operator error in failing to open the vent (if AC power is available) are not considered.

In summary, almost all the phenomena of interest to severe accident phenomenology are treated in the Perry CET. However, the submittal is not transparent on the treatment of phenomena such as combustion of carbon monoxide and recriticality of the degraded core after core damage.

2.1.2.3 Containment Failure Modes and Timings

The Perry Mark III steel containment has an internal design pressure of 15 psig. For the IPE submittal, the following ten potential containment failure modes were identified by the containment architect (Gilbert/Commonwealth):

1. Dome knuckle
2. Dome apex
3. Cylinder
4. Personnel airlock
5. Equipment hatch

6-8. Penetrations P123, P205 and P414

9-10. Steel and concrete anchorages

The most likely failure locations were found to be the containment penetrations. Of all the containment failure modes, these penetrations have the lowest failure pressure (57-60 psig). Two unlikely, but important, failure locations that were identified in the analysis are the steel and concrete anchorages, which can in turn fail the RPV injection lines and cause the loss of suppression pool water inventory. Anchorage failure can also lead to releases from the drywell directly to the environment without suppression pool scrubbing. A median failure pressure of 64.3 psig was arrived at for the containment failure pressure. The conditional probability of failure of the containment anchorages was also evaluated, as was the effect of containment overpressure failure upon RPV injection.

The internal design pressure of the Perry drywell is 30 psig. The following five failure modes were identified for the drywell:

1. Drywell wall
2. Drywell roof
3. Drywell head
4. Drywell equipment hatches
5. Drywell personnel airlock

The failure of the drywell head was found to be the most significant mode of failure. The drywell was found to have a median failure pressure of 70.8 psig. However, it should be noted here that in the evaluation of drywell capacities, the results obtained from plant specific calculations were discarded and data for Kuosheng plant is used. The reason for using the Kuosheng data is stated to be the geometric similarity of the two plants.

The effects of high temperature on concrete, reinforcing steel, steel plate, and seal materials are addressed in Appendix H.1 of the submittal. The only significant effects were noted to be for the steel plates and seal materials. Data for the high temperature performance of the drywell hatch seals and electrical penetration seals are provided in Section 8 of Appendix H.1 for some cases, and referenced there for others. However, it is not clear how these data are used in the actual evaluation of containment fragilities at high temperatures in the IPE submittal. The leakage behavior of inflatable seals at elevated temperatures (such as the ones found in the personnel lock and escape lock) has been a subject of study in the recent past, and an analytical/empirical methodology is reported in Reference [4] for the evaluation of threshold pressure of leakage for these seals. The licensee should include such data for the evaluation of the containment fragilities at elevated temperatures.

In summary, the evaluation of the containment capacities in the submittal appears to be acceptable. However, the use of data for the Kuosheng plant drywell capacities for the Perry plant, and the lack of clarity on how the effect of elevated temperatures upon the containment capacities is treated are two shortcomings of the submittal.

2.1.2.4 Radionuclide Release Categories and Characterization

The results of CET analyses lead to an extensive number of end-states, which are in turn binned for source term analyses. This process is analogous to the definition of PDSs for the level 1 to level 2 interface. Outcomes of the CETs are classified into a manageable number of releases, which are characterized by similarities in accident progression and source term characteristics.

The Perry IPE submittal [1] defined 25 source term release categories, including intact containment/recovered states (Figure 4.7.1-1 of the submittal). The grouping logic is also described in that figure.

The Perry IPE source term bins are defined based on the following combination of binning characteristics:

1. Containment Bypass: Describes whether containment bypass occurs due to interfacing systems LOCA (V sequence), and main steam line breaks.

2. Debris Coolability In-vessel: Provides an indication of whether or not ex-vessel debris interactions are possible.

3. Containment status at core damage Indicates whether or not the containment is isolated at the point core damage begins.

4. Time of Containment Failure: This indicator differentiates the time of containment failure (for non-bypass sequences and sequences that involve no failure prior to core damage) into Early (prior to or at vessel breach), Late (several hours after vessel breach), and No failure.

5. Mode of Containment Failure The modes of containment failure are anchorage failure, penetration failure, and containment venting.

6. Suppression Pool Bypass Suppression pool bypass is classified into early, late and no bypass for all sequences.

7. Containment Spray Operation This indicator classifies sequences with pool bypass based on whether or not sprays operate over the entire time period when radionuclide release is occurring.

8. Type of Core-Concrete Interaction Three types of core concrete interactions were considered, namely, dry, wet, and no CCI. The wet cases were binned with the dry CCI sequences.

The determination of the source term magnitudes, composition, and release timing for each release category was performed using the MAAP 3.0B code. MAAP calculations were then performed for 11 of these release categories, and the releases for the other categories were characterized by similarity to one of the calculated source terms. The representative accident sequences used for the release category calculation were selected on the basis of a significant frequency of occurrence at the PDS level and a significant probability of containment failure. The accident sequences selected for the specification of the release fraction corresponding to each source term category are listed in Table 4.7.3-1 of the submittal. The timing of the releases for each release category are provided in Table 4.7.3-2. The actual magnitude of the source terms for each category are listed in Table 4.7.3-3. Recommendations for source terms for the unanalyzed source term categories, based on the analyzed sequences, are provided in Table 4.7.3-4. From Table 4.7.4-2, release categories that have frequencies greater than 1.0E-10 per reactor-year appear to have been reported in the submittal, although the IPE submittal does not report the cutoff value. Generic Letter 88-20 states that "any functional sequence that has a core damage frequency greater than or equal to 10^{-6} per reactor year and that leads to containment failure which can result in a radioactive release magnitude greater than or equal to the BWR-3 or PWR-4 release categories of WASH-1400" should be reported. The IPE submittal fulfills this requirement.

Other than the magnitude of the releases of the fission products, no other characteristics are listed in the submittal. This audit would have been facilitated if characteristics such as time of release, duration of release, energy of release, height of release, and isotopic fractions released to the environment had been provided in the submittal. Though the Generic Letter does not require listing of these characteristics; nevertheless, they are equally important in terms of providing insights about containment vulnerabilities.

A cursory comparison of Table 4.7.3-3 of the IPE submittal and Tables 3.3-1 to 3.3-11 of NUREG/CR-4551 [2] indicate that the source terms reported in the submittal for the volatiles and noble gases are of the same order of magnitude, while those of Te (assumed to be present as TeO_2 in the submittal) are larger in Perry. The releases for other species could not be compared since NUREG-1150 reports an inordinately large number of release bins. In summary, the source terms reported in the IPE submittal are sufficient and satisfy the scope of the generic letter. It is noted, however, that all the source term calculations in the submittal were performed for transients, and no calculations were performed for (critical) ATWS sequences, which can lead to large releases.

2.1.3 Quantitative Core Damage Estimate

2.1.3.1 Severe Accident Progression

The IPE submittal cites the use of the MAAP 3.0B BWR version 7.02 code for the analysis of accident progression and fission product source terms. The MAAP input file is provided in Appendix H.5. It is stated in Section 4.6 that information regarding the timing of key events, containment loads, mitigation effects of injection systems, the generation and combustion of CO and H_2, and core concrete interactions were obtained from the MAAP calculations. However, no other details are available regarding the accident sequences for which MAAP analyses were performed and the results of the performed MAAP analyses. For evaluating the source terms, MAAP calculations were performed for 11 accident sequences, and those transients are listed in Table 4.7.3-1. Some results for key timings of these accident sequences are listed in Table 4.7.3-2.

Sensitivity calculations are stated to have been performed on the following parameters.

1. The coefficient for critical heat flux used in debris coolability.
2. Containment failure area.
3. Impact of core geometry upon cladding oxidation.
4. Delayed hydrogen combustion, and
5. Diffusion flame modelling during ignitor operation.

Appendix B of Reference [3] recommends sensitivity studies on more than 20 parameters for the MAAP 3.0 BWR code. It is surprising that one of the authors of that reference has performed the MAAP calculations for this submittal and has not taken into consideration his own recommendations.

Phenomenological uncertainties have received a limited treatment in the IPE submittal. Most of the phenomena of importance to BWR severe accident phenomenology have been treated in the submittal, and they include the following:

(1) Hydrogen combustion,
(2) Direct containment heating,
(3) Steam explosions (both alpha mode and pedestal cavity),
(4) Molten core-concrete attack, and
(5) Containment and drywell pressurization due to blowdown and vessel breach.

The phenomena not treated in the Perry IPE submittal include the following:

(1) Core recriticality due to loss of control materials, and
(2) Combustion of carbon monoxide generated due to MCCI.

The lack of detailed MAAP calculated results, and a lack of sensitivity calculations (on those sensitivity parameters recommended in Reference [3]) are two shortcomings of the submittal.

2.1.3.2 Dominant Contributors: Consistency with IPE Insights

Table 1 shows a comparison of the conditional probabilities of the containment failure modes set out in the Perry IPE submittal, together with the results of the NUREG-1150 study [2] for the Grand Gulf plant. All comparisons are made for internal initiating events only.

The Perry core damage frequency from internal events in the submittal is an order of magnitude larger than that reported in the Grand Gulf NUREG-1150 report. Nearly 32.5% of the Perry core damage frequency is contributed by sequences which involve containment failure at the time of core damage. These sequences lead to early containment failure or venting at the time of core damage. Another 62% of the core damage frequency is contributed by transients with containment heat rejection available, and which have a depressurized reactor vessel. These sequences contribute to less than 1% to the core damage frequency of the Grand Gulf Plant. Finally, station blackout sequences contribute about 97% of the CDF (3.96×10^{-6} per reactor year) in Grand Gulf; while in Perry, they account for 4.4% of the CDF (5.6×10^{-7} per reactor year). With such large differences in the CDF, and in the dominant accident sequences, it is to be anticipated that the containment response will be altogether different in the two plants. Only a cursory comparison can be made between the two plants. It appears that all the station blackout sequences lead to some form of containment failure in Perry, while for Grand Gulf, venting

Table 1 Containment Failure as a Percentage of Internal Events CDF: Comparison with Other PRA Studies

Containment Failure Mode	Perry	Grand Gulf
Early Failure with Early Pool Bypass, No Spray	3.5	15.8
Early Failure with Early Pool Bypass, with Spray	0.5	4.9
Early Failure with Late Suppression Pool Bypass	12.5	0.7
Early Failure with No Suppression Pool Bypass	8.3	21.8
Late Failure	7.4	28.4
Venting	29.3	3.8
Intact	39.1	23.0
Core Damage Frequency, yr^{-1}	$1.3x10^{-5}$	$4.1x10^{-6}$

followed by late containment failure was found to be the dominant contributor to containment failure. In Perry all the ATWS sequences seem to lead to early containment failure and venting, while in Grand Gulf, there is a 6% chance of no containment failure, and a 7% chance of late containment failure. Beyond these simplified comparisons, no other conclusions could be reached, since submittal Tables 4.3.3-1 and 4.5.3-3 do not list all the sequences that contribute to all the PDSs.

2.1.3.3 Characterization of Containment Performance

By performing the MAAP calculations, the IPE team assessed the containment loading and characterized containment performance for important accident sequences. As already mentioned in Section 2.1.3.1, a few sensitivity studies were performed using the MAAP code. However, there are some reservations about the treatment of uncertainties in the phenomena and their impact upon the containment performance. The uncertainties in the phenomenology are treated indirectly. In the base case IPE quantification, the analysis is performed within the MAAP framework of treatment of the phenomenology. Single point estimates are obtained for the important phenomenological questions and for containment loads, and the containment loads are compared with the containment fragility curves to arrive at the probability of drywell/containment failure. Subsequently in Section 4.6.2, sensitivity analyses are performed upon the CET analyses. For each event node considered to be a source of uncertainty, sensitivity studies were performed by varying the split fraction of the node between its logical limits. The results of performing these sensitivity studies upon the CET outcomes are reported in Section 4.6.2. Within this framework of treatment, the phenomenological uncertainties are explored. Key phenomenological sensitivities treated here include the following: RPV depressurization during core damage, in-vessel coolability, mode of RHR spray operation, fraction of in-vessel zirconium oxidation, small burns, containment failure pressure, mode of containment failure, mode of in-vessel steam explosion bottom head failure, RPV lower head failure size, pedestal failure due to steam explosion, core-coolability ex-vessel, and human action to activate the hydrogen ignitor system. Of these sensitivities, only two were found to have an impact upon containment failure, namely, in-vessel coolability, and the probability of in-vessel steam explosion induced bottom head failure.

The treatment of containment failure in the submittal is adequate and follows the NUREG-1150 method of treatment of containment failure by comparing the pressurization rate with the containment fragility curves and estimating the probabilities of leak and gross rupture. However, the uncertainties associated with containment loading for different phenomena were not directly treated. Also, the effect of high temperature upon the containment failure pressure in the late phase of the accident is not treated in the submittal.

2.1.3.4 Impact on Equipment Behavior

The impact of the severe accident phenomena upon equipment behavior is treated in one case, where the effect of the containment overpressure failure upon RPV injection is considered in the containment event tree. The effect of containment failure upon the operating injection systems was evaluated by considering the following phenomena:

1. Injection system piping deformation,
2. Containment failure mode, and
3. Injection system degradation due to environmental conditions.

Similarly, the effect of containment failure upon the RHR containment spray was also evaluated. However, the possibility of the equipment failure at elevated temperatures prior to containment failure was not evaluated in the submittal.

2.1.4 Reducing the Probability of Core Damage or Fission Product Release

2.1.4.1 Containment Vulnerabilities, Safety Enhancements and Plant Modifications

A discussion of vulnerabilities is presented in Section 3.4.2 of the submittal, though this discussion corresponds mainly to the front-end portion of the IPE. A general discussion of the vulnerabilities of the Mark-III containment is provided in Section 3.4.2.3. Unique containment features are discussed in Section 6.1.2. It is reported in that section that one unique containment feature at Perry is the possible anchorage failure given containment failure, which can affect injection sources and lead to a loss of suppression pool water. No specific change in containment design is proposed; however, a number of procedural changes and several system modifications of interest to the front-end analysis are proposed.

2.1.5 Responses to The Recommendations of The CPI Program

As part of the Containment Performance Improvement (CPI) program, several design considerations for BWRs with Mark III containments were considered in the IPE submittal (Section 4.6.3). These include the recommendations of the CPI program, and, in some cases, are system modifications identified as a part of the submittal. The following containment and system improvements were considered:

1. Passive containment venting design.

2. ATWS alternate shutdown and ADS inhibit design consideration.

3. Backup power to hydrogen ignitors for operation under station blackout conditions and

4. Combinations of the above.

A passive vent design would include a rupture disk in an alternate vent line to open automatically upon containment overpressure. Although the alternate vent path has not been designed, the associated change in the containment failure probabilities was evaluated in the submittal, and the results displayed in Table 4.6.3-1. A substantial decrease in the probability of RPV failure and containment failure was observed.

The ATWS alternate shutdown and ADS inhibit design was considered to address the critical ATWS sequences. An automatic ADS inhibit was found to reduce core damage frequency. The containment performance improvement includes a modification in the EOPs to control the RPV power level as a function of containment pressure, using a water level band such that the steam generated was within the containment vent heat removal capacity. The effect of this procedural and system modification was found to be a reduction of 13% to the conditional probabilities of both RPV failure and containment failure.

The provision of a backup power supply to the hydrogen ignitors was suggested by the CPI program to control hydrogen burning in station blackout sequences. However, since station blackout sequences were found to contribute less than 5% of the core damage frequency in the Perry IPE submittal, this provision would have limited impact on containment performance. Calculations performed as a part of the submittal show a 6% reduction of the conditional probability of containment structural failure, and 2% reduction of the conditional probability of RPV failure.

Finally, the joint effect of the passive vent design and ATWS alternate shutdown and ADS inhibit design change were evaluated. An increased chance of core damage arrest, and a reduction in core damage frequency and conditional probability of containment failure, were found to result. The effect of combined passive vent design, ATWS alternate shutdown and ADS inhibit design, and the provision of backup power to the hydrogen ignitors, was also investigated. As explained earlier, the additional effect of adding the backup power supply to the HIS system was found to be insubstantial.

The effect of these possible containment design changes upon the source terms was not evaluated in the submittal.

2.2 IPE Strengths and Weaknesses

2.2.1 Strengths of IPE

1. The Back-End portion of the IPE is complete with regard to the information provided in the subject areas identified in Generic Letter 88-20. However, some clarifications and detailed results are needed in the submittal to confirm the IPE findings.

2. For the most part, the separate models used in the Perry IPE Back-End analysis are technically sound.

3. The Perry IPE employs a simplified version of the NUREG-1150 PRA methodology. It should be noted that the methodology used in the IPE submittal is more easily understandable than that used in the NUREG-1150 PRA, without sacrificing any of the important details.

4. The Perry IPE treats most of the phenomena of importance in severe accident progression in accordance with Appendix I of the Generic Letter.

5. The containment event tree used in the IPE submittal is a condensed version of the event tree used in NUREG-1150 for the Grand Gulf plant. Again it should be noted that the smaller size of the tree and the presentation in Figure 4.5.2.1-1 make it more transparent than the Grand Gulf tree, while virtually all important top event nodes are included in the Perry CET.

6. The evaluation of containment capacities and the containment failure modes are detailed and complete.

7. The source term analysis in the Perry IPE submittal has been performed with state-of-the-art tools, and sufficient source terms have been calculated.

2.2.2 Weaknesses of IPE

1. Involvement of CEI personnel in the IPE appears to have been limited, although the task breakup indicate at least one CEI personnel for each task. The majority of the modelling and analysis effort appears to have been performed by outside contractor personnel (NUS corporation, Gilbert Commonwealth, and Gabor, Kenton and Associates, Inc.).

2. The absence of results from MAAP analyses of accident sequences, and the lack of sensitivity analyses of these accident sequences (at a minimum, those suggested in Reference 5), are important shortcomings of the submittal.

3. The treatment of the containment capacity at elevated temperatures, though discussed in Appendix H, is weak.

4. The treatment of uncertainties for different phenomena is a shortcoming of this IPE submittal. Phenomenological uncertainties associated with the containment loading in BWR accident sequence modelling are not treated in the Perry IPE submittal; however, a number of sensitivity analyses were performed.

2.3 Evaluation of Principal Issues from Licensee Response to NRC Questions

Following the issuance of the draft version of this TER, the NRC forwarded a number of questions to the licensee. The licensee's responses to these questions [5] have been reviewed, and generally found to be acceptable. A summary of the principal issues addressed in the licensee's response to the NRC questions is provided below.

B.E.1 Recriticality After Core Damage Due to Meltdown of Control Rods

For non-ATWS sequences, during core damage progression after meltdown of control rods, there is a (small) possibility of core becoming recritical. This was not addressed in the Perry (and several other) IPE submittals. The response claims to have addressed this question implicitly as a part of the late containment failure (due to steam generation) event. Since there is no dependency on any recriticality node, we can only guess that the analysts conservatively increased the probability of containment failure by overpressurization to account for the recriticality event. However, it is not anticipated that recriticality for non-ATWS sequences will significantly add to the conditional probability of containment failure, and it appears that the licensee is aware of this issue.

CO Generation

Although not mentioned in the submittal, the licensee has used the NUREG-1150 analyses for Grand Gulf for:

(a) moles of CO generated for dry and wet pedestal floor (although not all the Grand Gulf CET branches appears to have been included), and indirectly for drywell pressure, and

(b) AICC burn code used in the NUREG-1150 code was used to calculate the burn pressures.

The licensee used the CO generation and combustion results from the NUREG-1150 analyses, whereas from the submittal, it appeared that the MAAP results had been used for the late phase of the accident. Hence, the treatment of CO generation and combustion appears to be comparable to NUREG-1150 analyses.

B.E.2 In-Vessel Steam Explosion

The licensee has concluded that the NUREG/CR-4551 estimates of the probability of IVSE are one order of magnitude too large. It appears that the necessary insights have been gained through their sensitivity analyses and a literature review to buttress the IPE results.

B.E.3 Drywell Fragility Curve

The response to this question is still not convincing. This question was raised because it appears that the Perry-specific calculation by Gilbert Commonwealth has been discarded by NUS. It is possible that the drywell of the Perry plant is very similar to Kuosheng, but the licensee has not provided proper justification for using the containment fragility curve for Perry in the submittal. It is also not clear to the reviewers why the containment capacity (64.3 psig) is about 10-20 psig larger than other BWRs with Mark-III containments (i.e., Grand Gulf).

B.E.4 Impact of Elevated Temperature on Equipment Performance

The response to this question implies that the only equipment that are considered (for equipment performance in back-end analyses) are Hydrogen Ignition System (HIS) and H_2 analyzer (for the HIS). Containment spray header plugging and RHR heat exchanger plugging was not considered after core damage. In addition, the effect of local hydrogen detonations within the containment (wetwell) upon the hydrogen ignition system (i.e., temperatures above 345 F) also does not appear to have been considered.

B.E.5 AC Power Recovery, Recovery of Sprays and Hydrogen/Carbon Monoxide Burn

It was not clear from the submittal if recovery of AC power late, RHR sprays, and hydrogen ignitor operation late in the accident where questioned. However, in retrospect, since the (abridged) Grand Gulf event tree was used, it appears that the same split fractions as in NUREG-1150 are used for the four late burn-related questions, i.e., for RHR spray operation and steam condensation, probability of late burn, late availability of AC power and hydrogen ignition sources.

In addition, for the late controlled burn (i.e., use of containment sprays to control burn pressure), the NUREG-1150 AICC burn model has been used for both the base case and the sensitivity study. There are no objections to this model, but, the use of the NUREG-1150 analyses and the description of the various subcases (as described in the response) should have been included in the submittal.

3. OVERALL EVALUATION AND CONCLUSIONS

A submittal-only review of the Perry IPE indicated that, for the most part, the IPE is transparent and well documented. Some details, such as the method of treatment of combustion of noncondensable gases, and the bases for the split fractions used for the quantification of some severe accident issues, are lacking. Following the issuance of the draft version of this TER, a number of questions were forwarded to the licensee. A review of the licensee reponses showed that the licensee had performed additional detailed analyses which were not documented in the submittal. Based upon the review of the submittal and the licensee responses to the back-end questions, it appears that the licensee has provided a substantial amount of *Back-End* information which contributes to the resolution of the severe accident vulnerability issues for the Perry nuclear plant.

4. REFERENCES

1. "Perry Nuclear Station Individual Plant Examination Submittal Report," prepared by Consolidated Edison Company, November 1991.

2. "Evaluation of Severe Accident Risks: Grand Gulf, Unit 1,", NUREG/CR-4551, SAND*^-1309, Vol. 6, Part 1.

3. Kenton, M. K., and Gabor, J. R., "Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP 3.0B," EPRI report.

4. Parks, M. B., "Leakage Behavior of Inflatable Seals Subject to Severe Accident Conditions," Nuclear Engineering and Design, 131, 175-186, 1991.

5. Letter from R. Stratman (Centerior Energy) to U.S. Nuclear Regulatory Commission, dated November 24, 1993.

APPENDIX

IPE EVALUATION AND DATA SUMMARY SHEET

PWR Back-End Facts

Plant Name

Perry

Containment Type

Mark III containment

Unique Containment Features

The possible anchorage failure given containment failure, which can affect injection sources and lead to a loss of suppression pool water (Applicable to all Mark III containments)

Unique Vessel Features

None found

Number of Plant Damage States

75 total; 15 used.

Containment Failure Pressure

64.3 psig (median); Drywell Failure Pressure = 70.8 psig (Median)

Additional Radionuclide Transport and Retention Structures

No additional structures credited

Conditional Probability That The Containment Is Not Isolated

0.0

Important Insights Including Unique Safety Features

None found related to back end

Implemented Plant improvements

Plant improvements under consideration include: (1) passive containment venting design, and (2) ATWS alternate shutdown and ADS inhibit design consideration

C-Matrix

See Table 4.5.3-2 of the submittal