ERI/NRC 95-103

TECHNICAL EVALUATION REPORT OF THE SHEARON HARRIS INDIVIDUAL PLANT EXAMINATION BACK-END SUBMITTAL

Final Report

May 1995

Energy Research, Inc.
P.O. Box 2034
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Prepared for:
SCIENTECH, Inc.
Rockville, Maryland

Under Contract NRC-04-91-068
With the United States Nuclear Regulatory Commission
Washington, D.C. 20555

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E. EXECUTIVE SUMMARY

This Technical Evaluation Report (TER) documents the findings from a review of the back-end portion of the Individual Plant Examination (IPE) for the Shearon Harris Nuclear Power Plant (SHNPP). The primary intent of the review is to ascertain whether or not, and to what extent, the back-end IPE submittal satisfies the major intent of Generic Letter (GL) 88-20 and achieves the four IPE sub-objectives. The review utilized both, the information provided in the IPE submittal, and additional information provided by the licensee, Carolina Power & Light (CP&L) Company as a part of the responses to NRC questions. The IPE was a joint utility-contractor effort, with SAROS and Gabor, Kenton, and Associates (GKA) being the contractors responsible for the back-end analyses. An internal peer review of the back-end portion of the submittal was performed by the IPE team. The CET development and quantification was performed by SAROS and reviewed by CP&L employees. MAAP analyses and quantification of releases were performed by CP&L employees, directed by GKA, and reviewed by SAROS. Special issues such as those beyond the scope of the MAAP code, were addressed by a white paper prepared by GKA, and reviewed by CP&L and SAROS. After the analyses were completed, an utility team was established to review the results, develop insights and develop potential modifications and enhancements. The team was composed of members from nuclear plant operations and corporate management who had experience in plant operation. However, no significant changes in the IPE resulted from these reviews.

The IPE submittal is consistent with the level of detail discussed in the "Submittal Guidance Document", NUREG-1335.

E.1 Plant Characterization

The Shearon Harris plant is a 3-loop Pressurized Water Reactor (PWR) of Westinghouse design. The following plant-specific features are important for accident progression in the Shearon Harris plant:

- The cavity in the Shearon Harris plant is composed of a cylindrical region below the RPV connected to a rectangular tunnel by a normally open keyway. The cavity floor is 12 feet thick, covered with a 0.25 inch liner, which is in turn covered with 1.5 ft of concrete. The horizontal, rectangular section of cavity is 16 ft X 12 ft in cross-section and communicates directly to the lower compartment.

- The design of the secondary shield wall (the wall between the steam generator compartments and the containment wall, located 19 feet from the containment wall) leads to the possibility of debris (leaving the cavity) flow to the annular compartment surrounding the secondary shield wall, and directly contacting the liner.

- The large floor area of the cavity allows significant water accumulation without submergence of the RPV lower head. Even if the entire inventory of the RWST is injected onto the containment floor, the water level in the Shearon Harris plant will still

be two feet below the RPV bottom head. The cavity is also surrounded by a 1.5 ft concrete curb, which means that water will overflow into the cavity only if RWST inventory is injected into the containment. Therefore, lower vessel head cooling could not be credited.

- The concrete used in the Shearon Harris plant is stated to be a quartz-based aggregate. This type is similar to basaltic concrete, and will lead to generation of smaller quantities of non-condensible gases due to core-concrete interactions, as compared to limestone concrete.

However, in several other aspects such as RCS water volume, power, and core inventory, the Shearon Harris plant is similar to the Surry and Zion plants.

E.2 Licensee's IPE Process

The IPE methodology employed in the Shearon Harris IPE submittal for the back-end evaluation is clearly described, and the IPE is logical and consistent with GL 88-20. The front end analyses show that the overall CDF (including contributions from internal events and flooding) is 7×10^{-5} per reactor year, and is not dominated by any single initiating event. Small LOCAs and loss of offsite power sequences contribute to more than 60% of the total CDF. The outcome of the front end analyses were grouped into core damage bins. A Containment Safeguard Event Tree (CSET) was then developed to interface the core damage bins with the containment safeguard functions. The binning of the outcomes of the CSET leads to the quantification of the Plant Damage States (PDSs).

Core Damage Bins (CDBS) are a combination of five separate binning characteristics, i.e.,

1. RCS leakage rate,
2. RCS pressure at vessel breach,
3. Reactor cavity status,
4. Availability of heat removal by steam generators, and
5. Time of RPV failure.

There are 96 possible CDBs; however, some of the combinations (for instance, large LOCAs and high RCS pressure) are illogical. Only 17 bins were found to remain after weeding out the illogical combinations.

The CSET is a small event tree comprised of six top events, each of which is further developed by fault trees. The top events of the CSET are the following:

1. Containment Sprays Functional in the Injection Mode (1 out of 2 trains),
2. Containment Fan Coolers Functional (2 out of 4 fan coolers),
3. Containment Fan Coolers Functional (4 out of 4 fan coolers),
4. Containment Sprays Functional in Recirculation Mode,

5. Containment Isolation Failure (Large Hole Size), and
6. Containment Isolation Failure (Small Hole Size).

There are 18 possible outcomes for a CSET for each core damage bin, and there are 17 possible core damage bins. Hence 306 PDSs are possible. However, to manage this process, the PDSs were truncated by using a cut-off frequency of 10^{-7} per reactor year. After the truncation, a total of 17 PDSs were found to result.

Probabilistic quantification of severe accident progression involved the development of a small Containment Event Tree (CET), and use of supporting fault trees to evaluate the split fractions for each CET node. The CET contains the following fifteen top events:

(1) In-vessel recovery successful,
(2) Hydrogen burn after in-vessel recovery,
(3) Containment not bypassed,
(4) Successful containment isolation,
(5) Containment leakage rate is small,
(6) Early containment overpressure is prevented,
(7) Containment liner intact,
(8) Ex-vessel cooling successful,
(9) Late hydrogen burn mitigated,
(10) No late containment failure (steam overpressure),
(11) No late containment failure (non-condensible gas),
(12) No basemat failure,
(13) Late revaporization does not occur,
(14) Radionuclide scrubbed by overlying pool, and
(15) Radionuclide scrubbing by containment sprays.

The same CET is quantified for all PDSs. The CET analyses were performed using a small event tree/large fault tree methodology. The CET and the fault tree analyses are transparent and treat all the phenomena of interest to severe accident progress in a PWR with a large, dry containment.

The results of the CET analyses lead to an extensive number of end-states, which are classified into a manageable number of release categories, characterized by similarities in accident progression and source term characteristics. The main characteristics of the CET end-states considered when developing these release categories in the submittal were:

- Release energy,
- Containment isolation failure size,
- Time of failure, and
- Isotopic composition

plant does not permit communication between the cavity and the upper compartment. Therefore, the submittal assumes that only a small fraction of the core debris ejected at high pressure during vessel breach will be transported to the lower compartment of the containment and participate in direct containment heating. Secondly, the Shearon Harris containment was determined to have a large capacity (i.e., the median containment failure pressure is calculated to be 153 psig).

The small probability of late containment failure is attributed to the following two reasons. First, the containment has a large cavity floor area, and therefore the submittal assumes that there is a high conditional probability of coolability of debris on the cavity floor by an overlying pool of water. Secondly, the concrete type in the Shearon Harris plant is a quartz-based aggregate, which is similar to Basaltic concrete. The generation of non-condensible gases were found to be very small for this type of concrete, and hence the conditional probability of late overpressure failure is calculated to be low.

The submittal indicates a high conditional probability of releases due to induced steam generator tube rupture events. The EOPs for the Shearon Harris plant require the operators to restart the Reactor Coolant Pumps (RCPs) if they are available, when there is inadequate core cooling.

Table E.1 Containment Failure as a Percentage of Total CDF: Comparison With Other PRA Studies

Containment Failure Mode	Shearon Harris IPE	Surry NUREG-1150	Zion NUREG-1150
Containment Failure with In-Vessel Recovery (Prior to Vessel Breach)	3.2	-**	-**
Early Failure	0.25	0.7	0.5
Late Failure	1.0	5.9	24.0
Very Late Failure	3.6	NA+	NA+
Bypass (V)	0.7	7.6	0.2
Bypass (SGTR)	6.5	4.6	0.3
Isolation Failure	0.3	NA++	1.0
Intact	84.5	81.2	73.0
Core Damage Frequency, yr^{-1}	$7x10^{-5}$*	$4.1x10^{-5}$	$6.2x10^{-5}$

* Includes Flooding
\+ Included as a Part of Early Containment Failure
\+ Included as a Part of Late Containment Failure
** Not Applicable

This results in clearing of the RCP loop seals, and establishes a path for natural circulation to the steam generator tubes. Accordingly, there is a probability of induced SGTR due to natural circulation of hot gases. The induced SGTR provides a means of containment bypass, which in the absence of steam generator secondary side water inventory, will lead to unscrubbed fission product releases to the environment. These releases are much larger if cycling of the secondary side SRVs caused the valves to become stuck open. The submittal has identified this mode of induced containment bypass as the principal contributor to radiological releases in the Shearon Harris plant.

The submittal makes use of the MAAP code to calculate the radiological releases. The releases are dominated by steam generator tube rupture sequences, through a cycling or stuck-open SRV. The bypass sequences (SGTRs and ISLOCA) contribute to 67% of the release frequency. The releases calculated for these sequences are large, and it can be said that the risk profile for the Shearon Harris plant is dominated by the releases from the containment bypass sequences.

E.4 Generic Issues and Containment Performance Improvement (CPI) Issues

One of the recommendations of the CPI program pertaining to PWRs with large dry containments was that the utility should evaluate the IPE results for containment and equipment vulnerabilities to hydrogen combustion (local and global), and point out any need for procedural and/or hardware improvements. The submittal documentation does not explicitly discuss the recommendations of the CPI program. However, in response to the NRC questions, the licensee discussed the recommendations of the CPI program, and their treatment of the CPI program recommendations [8]. The licensee has studied the impact of local and global hydrogen combustion upon containment and equipment performance at Shearon Harris. However, it was stated that no recommendations could be identified for improving containment performance relevant to global hydrogen combustion at SHNPP. With regards to the localized hydrogen combustion, a quantitative assessment of hydrogen release points (pressurizer relief tank, hot and cold legs, etc.) by the licensee failed to identify any enclosed spaces that would allow for hydrogen buildup and combustion that can endanger equipment. In addition, it was noted that most of the safety grade equipment is located outside the containment, and is not impacted by adverse environment inside the containment. The only important systems located inside the containment are the containment sprays and fan coolers. The licensee stated that the spray nozzles are reasonably insensitive to hydrogen combustion, but the fan coolers have the potential to be impacted by a local hydrogen burn. The likelihood of the failure of fans was quantitatively analyzed, and found to be rather insignificant. Two factors helped to reduce the significance of local hydrogen combustion. First, many accident sequences (that lead to core damage) involve failure of support systems, and fan coolers were not functional during accident progression. For sequences that had fan coolers functional, but unable to operate (i.e, due to unavailability of AC power), recovery of AC power caused both fan coolers and sprays to operate. With either system being sufficient for containment heat removal, the fan coolers can fail without adversely impacting accident progression. Hence, the overall IPE results for containment failure and the radionuclide release frequency were not found to be impacted by the failure of fan coolers due to local hydrogen combustion. In addition, the licensee noted that a

review of the results from severe accident analyses did not identify any volumes inside the containment where a sufficient buildup of hydrogen could lead to Deflagration-to-Detonation Transition (DDT). However, the licensee stated that a containment walkdown has not been performed to identify passages conducive to the occurrence of DDT. In summary, all the CPI recommendations are addressed by the licensee.

E.5 Vulnerabilities and Plant Improvements

The submittal does not define "vulnerability", particularly as related to containment analyses. However, it should be noted that a few initiators, particularly the loss of offsite power and the small break LOCA dominate the CDF profile. In addition, it should also be noted that the EOP requiring the operators to restart the RCP to provide additional cooling during the course of severe accidents was found to have important consequences. In the submittal, this operator action is shown to lead to induced SGTR and thereby large radiological releases. In response to an NRC question on the induced SGTR (due to reactor coolant pump restart), the licensee stated that this failure mode of steam generator tubes is not unique to the Shearon Harris plant, and that the issue is being addressed by the Westinghouse Owners Group. It should be noted that this potential failure mode has not been identified by other IPEs reviewed by ERI, for other Westinghouse PWRs.

A multi-disciplinary utility team was formed to review the IPE results and suggest plant modifications and improvements. Although this team made several suggestions for improvements to the plant, no improvements based on the containment analyses were considered necessary.

E.6 Observations

The assessment of this review is that the Shearon Harris IPE submittal documentation, and the responses to the NRC review team questions, contains substantial *Back-End* information regarding the severe accident vulnerability issues for the Shearon Harris plant.

The following are the major findings of the Shearon Harris IPE submittal:

- The Shearon Harris submittal shows that the overall CDF of 7×10^{-5} per reactor year is not dominated by any single initiating event. Small LOCAs and loss of offsite power sequences contribute to more than 60% of the total CDF.

- The containment analyses indicate that there is a 15% conditional probability of releases, and 85% conditional probability of intact containment.

- Containment features such as large cavity floor area, limited communication between the cavity and the containment, and the larger calculated containment capacity, all contribute to the low conditional probability of containment failure.

- The releases for the Shearon Harris plant are dominated by (pre-existing and induced) steam generator tube rupture sequences.

The important points of the technical evaluation of the Shearon Harris IPE back-end analysis are summarized below:

- The Back-End portion of the IPE supplies a substantial amount of information with regards to the subject areas identified in Generic Letter 88-20.

- The Shearon Harris IPE provides an evaluation of all phenomena of importance to severe accident progression in accordance with Appendix I of the Generic Letter.

- The licensee has addressed the recommendations of the CPI program [8].

The "strengths" of the IPE include the following:

- Phenomena that lead to early containment failure (i.e., direct heating, hydrogen combustion) have been treated in considerable detail. The licensee has attempted to quantify the uncertainty in phenomena such as Direct Containment Heating (DCH) and hydrogen combustion. The submittal has also identified a plant-specific containment failure mode that involves direct liner attack by high pressure-induced dispersal of core debris, and quantified the probability of liner failure due to direct contact by core debris. The results of the containment failure analysis show that early containment failure for the Shearon Harris plant is dominated by hydrogen combustion.

- The licensee has evaluated the positive and negative consequences of operator actions to add water and cool a degraded core. Additional hydrogen generation by core reflood, possibility of in-vessel steam explosions, and thermally-induced steam generator tube rupture by RCP restart, have all been treated in the submittal.

- The submittal has identified one specific procedural action that exacerbates the outcome of certain severe accident sequences. A procedure that requires the operator to restart the RCP to provide core cooling was found to lead to thermally-induced steam generator tube rupture.

A minor weakness in the submittal pertains to the quantification of basic events in the fault trees. A number of assumptions are made in the quantification of basic events, and in some cases, unsupported values are assigned to the basic events. However, these basic events are not expected to have a large impact on the overall results for containment failure and radionuclide releases.

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NOMENCLATURE

ATWS	Anticipated Transient Without Scram
CDB	Core Damage Bin
CDF	Core Damage Frequency
CET	Containment Event Tree
CSET	Containment Safeguards Event Tree
CHR	Containment Heat Rejection
CPI	Containment Performance Improvement
CP&L	Carolina Power and Light, Inc.
DCH	Direct Containment Heating
DF	Decontamination Factor
ECCS	Emergency Core Cooling Systems
EOP	Emergency Operating Procedure
EPA	Electrical Penetration Assembly
EPRI	Electric Power Research Institute
EQE	Equipment Qualification Engineers, Inc.
ERI	Energy Research, Inc.
ESF	Engineered Safety Features
EVSE	Ex-Vessel Steam Explosion
GE	General Electric
GKA	Gabor, Kenton, and Associates, Inc.
GL	Generic Letter
IPE	Individual Plant Examination
ISLOCA	Interfacing Systems Loss of Coolant Accident
IVSE	In-Vessel Steam Explosion
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
MAAP	Modular Accident Analysis Program
MCCI	Molten Core Concrete Interactions
MSIV	Main Steam Isolation Valve
NRC	Nuclear Regulatory Commission
PDS	Plant Damage State
PRA	Probabilistic Risk Assessment
RC	Release Category
RCS	Reactor Coolant System
RHR	Residual Heat Rejection
RPV	Reactor Pressure Vessel
RWST	Reactor Water Storage Tank
SGTR	Steam Generator Tube Rupture
SHNPP	Shearon Harris Nuclear Power Plant
SRV	Safety Relief Valve
TER	Technical Evaluation Report
VB	Vessel Breach

1. INTRODUCTION

This Technical Evaluation Report (TER) documents the results of a review of the Shearon Harris Individual Plant Examination (IPE) Back-End submittal [1]. This TER complies with the requirements for IPE back-end reviews of the U.S. Nuclear Regulatory Commission (NRC) in its contractor task orders, and adopts the NRC review objectives, which 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 if the IPE submittal meets the intent of the Generic Letter 88-20, and
- To complete the IPE Evaluation Data Summary Sheet.

This TER complies with the requirements of the contractor task order for review. The remainder of Section 1 of this report describes the technical evaluation process employed in this review, and presents a summary of the important characteristics of the Shearon Harris nuclear power plant related to containment behavior and post-core-damage severe accident progression, as derived from the IPE. Section 2 summarizes the review technical findings, and briefly describes the submittal scope as it pertains to the work requirements. Each portion of Section 2 corresponds to a specific work requirement as outlined in the NRC contractor task order. A summary of the overall IPE evaluation, identification of IPE submittal strengths and weaknesses, and review conclusions are summarized in Section 3. Section 4 contains a list of cited references. Appendix A to this report contains the IPE evaluation summary sheets.

1.1 Review Process

The technical review process for back-end analysis consists of a complete examination of Sections 1, 2, and 4 to 7 of the IPE submittal. In this examination, key findings are noted; inputs, methods, and results are reviewed, and any issues or concerns pertaining to the submittal were identified. The primary intent of the review is to ascertain whether or not, and to what extent, the back-end IPE submittal satisfies the major intent of Generic Letter (GL) 88-20 [3] and achieves the four IPE sub-objectives. A draft TER based on the back-end portion of the submittal was submitted to the NRC in June 1994. A list of questions and requests for additional information were developed to help resolve issues and concerns noted in the examination process, and were forwarded to the licensee. The licensee responses [8] were reviewed. The final TER is based on the information contained in the IPE submittal [1] and the licensee responses to the RAIs [8].

1.2 Plant Characterization

The Shearon Harris Nuclear Power Plant (SHNPP) is a single unit, three-loop Westinghouse Pressurized Water Reactor (PWR), owned and operated by Carolina Power and Light (CP&L) Company. The rated thermal power of the plant is 2,775 MW.

A detailed description of the Shearon Harris containment and plant data are provided in Section 4.1 and Table 4.1 of the submittal. Figures 4-1 through 4-3 (of the submittal) illustrate some of the design features of the cavity and the containment that are important for severe accident progression.

The Shearon Harris containment building is a steel-lined concrete shell in the form of a vertical cylinder with a hemispherical dome and a flat base. The base is a (minimum of) 12 ft thick reinforced concrete slab, with a cylinder and dome resting on the top of the reinforced concrete slab. The containment is lined with a welded steel liner plate, the thickness of which varies from 0.25 inch on the basemat floor to 0.5 inch on the dome wall. The liner thickness is 0.375 inch on the cylinder wall.

The following plant-specific features are important for accident progression in the Shearon Harris plant:

- The cavity in the Shearon Harris plant is composed of a cylindrical region below the RPV connected to a rectangular tunnel by a normally open keyway. The cavity floor is 12 feet thick, covered with a 0.25 inch liner, which is in turn covered with 1.5 ft of concrete. The horizontal, rectangular section of cavity is 16 ft X 12 ft in cross-section and communicates directly to the lower compartment.

- The design of the secondary shield wall (the wall between the steam generator compartments and the containment wall, located 19 feet from the containment wall) leads to the possibility of debris (leaving the cavity) flow to the annular compartment surrounding the secondary shield wall, and directly contacting the liner.

- The large floor area of the cavity allows significant water accumulation without submergence of the RPV lower head. Even if the entire inventory of the RWST is injected onto the containment floor, the water level in the Shearon Harris plant will still be two feet below the RPV bottom head. The cavity is also surrounded by a 1.5 ft concrete curb, which means that water will overflow into the cavity only if RWST inventory is injected into the containment. Therefore, lower vessel head cooling could not be credited.

- The concrete used in the Shearon Harris plant is stated to be a quartz-based aggregate. This type is similar to basaltic concrete, and will lead to generation of smaller quantities of non-condensible gases due to core-concrete interactions, as compared to limestone concrete.

2. CONTRACTOR REVIEW FINDINGS

The present review compared the Shearon Harris IPE submittal to the intent of the Generic Letter (GL) 88-20, according to the guidance provided in NUREG-1335. The responses of the licensee were also reviewed. The findings of the present review are reported in this section, and follow the structure of Task Order Subtask 1.

2.1 Licensee's IPE Process

2.1.1 Completeness and Methodology

The IPE submittal is consistent with the level of detail discussed in the "Submittal Guidance Document", NUREG-1335.

The methodology employed in the Shearon Harris IPE submittal for the back-end evaluation is clearly described, and the IPE is logical and consistent with GL 88-20. The outcome of the front-end analyses are grouped into core damage bins. A Containment Safeguard Event Tree (CSET) was developed to interface the core damage bins with the containment safeguard functions. The binning of the outcomes of the CSET leads to the quantification of the Plant Damage States (PDSs). Probabilistic quantification of severe accident progression involved the development of a small Containment Event Tree (CET), and use of supporting fault trees to evaluate the split fractions for each CET node. The results of the CET analyses lead to an extensive number of end-states which are binned into release categories. The MAAP code is used to simulate the containment response and to quantify the source terms.

2.1.2 As-Built/As-Operated Status

The submittal states that the IPE models the containment and plant systems for the as-operated plant as of January 1, 1992. Insofar as the containment systems are concerned, it appears that all the Shearon Harris containment-specific features are modelled.

2.1.3 Licensee Participation and Peer Review of IPE

The IPE back-end analyses were performed by a team composed of CP&L employees, one contractor (SAROS) who developed the CET and performed the event tree analyses, and another contractor (GKA) who was responsible for deterministic severe accident analyses. An internal peer review of the back-end portion of the submittal was performed by the IPE team. The CET development and quantification was performed by SAROS and reviewed by CP&L employees. MAAP analyses and quantification of releases were performed by CP&L employees, directed by GKA, and reviewed by SAROS. Special issues such as those beyond the scope of the MAAP code, were addressed by a white paper prepared by GKA, and reviewed by CP&L and SAROS. After the analyses were completed, a team was established to review the results, develop insights and develop potential modifications and enhancements. The team was composed of members from nuclear plant operations and corporate management who had experience in plant operation.

2.2 Containment Analysis

The key characteristics of the Shearon Harris plant and containment were identified in Section 1.2. Tables 1 and 2 provide a summary comparison of the key design features of the Shearon Harris plant and containment systems with the Zion and Surry plants.

Table 1 Summary of Key Plant and Containment Design Features for the Shearon Harris Plant

Feature	Shearon Harris	Zion	Surry
Power Level, MW(t)	2,775	3,236	2,441
Volume of RCS Water, m^3	266	368	283
Free Volume of Containment, m^3	59,472	81,000	46,440
Mass of Fuel, kg	77,649	98,250	79,652
Mass of Zircalloy, kg	18,233	20,230	16,466
RCS Water Volume/Power, m^3/MW(t)	0.10	0.11	0.12
Containment Volume/Power, m^3/MW(t)	21.4	25.0	19.0
Zr Mass/Containment Volume, kg/m^3	0.31	0.25	0.35
Fuel Mass/Containment Volume, kg/m^3	1.31	1.21	1.72
Maximum Mass of H_2 due to Zirconium Oxidation, kg	799	886	721
Maximum H_2 Concentration, 10^{-3} moles/m^3	7	7	9

Table 2 Comparison of Containment Capacities

Feature	Shearon Harris	Zion	Surry
Design Pressure, psig	45	47	45
Mean Failure Pressure, psig	150	134	126
Concrete Aggregate	Quartz-based (similar to basaltic)	Limestone	Basaltic

In several aspects such as RCS water volume, power, and core inventory, the Shearon Harris plant is similar to the Surry and Zion plants, as shown in Table 1. Section 4.1 of the submittal provides a discussion of these plant-specific features.

2.2.1 Front-End/Back-End Dependencies

The outcome of the front end analyses were grouped into core damage bins. A Containment Safeguard Event Tree (CSET) was then developed to interface the core damage bins with the containment safeguard functions. The binning of the outcomes of the CSET leads to the quantification of the Plant Damage States (PDSs).

Core Damage Bins (CDBs) are a combination of five separate binning characteristics as described below:

1. *RCS leakage rate:* Indirectly describes the accident sequence. Four leakage rates were considered and they include the following:

   - Large break LOCA leakage rates
   - Medium break LOCA leakage rates
   - Small break LOCA leakage rates
   - Cycling valve relief rates

   It should be noted that the RCS leakage rates pertain to coolant loss at core damage. Thus, any change in leakage rates after core damage (i.e., loop seal LOCA after core damage in a station blackout sequence, hot leg rupture due to natural circulation, etc.) are treated by the CET, and will be described subsequently.

2. *RCS pressure at vessel breach:* Accident sequences are classified into three types depending on the pressure at vessel breach, namely, high (above 600 psig), medium (200 to 600 psig), and low (below 200 psig).

3. *Reactor cavity status:* Accident sequences are classified into two types, based on whether or not the RWST water is injected into the containment. The identified outcomes of this parameter are the following:

   - Wet (RWST injected into the containment)
   - Dry (i.e., minimal water present in the cavity)

4. *Availability of heat removal by steam generators:* Two states are described by this parameter:

   - Steam generator cooling functional
   - Steam generator cooling failed

5. *Time of RPV failure:* The time available between accident sequence initiation and vessel breach is described by this parameter. There are two possible choices for this parameter:

- Early vessel breach (within 4 hours from the time of accident sequence initiation)
- Late vessel breach (beyond 4 hours from the time of accident sequence initiation)

There are 96 possible CDBs; however, some of the combinations (for instance, large LOCAs and high RCS pressure) are illogical. Only 17 bins were found to remain after weeding out the illogical combinations. The bins are numbered from 1 to 17, and defined in Table 4-8 (page 4-63) of the submittal. Tables 4-9 through 4-14 (pages 4-64 through 4-69) of the submittal show the actual binning process for each accident sequence considered in the front-end analyses.

The CSET is a small event tree comprised of six top events, each of which is further developed by fault trees. The top events of the CSET are the following:

1. Containment Sprays Functional in the Injection Mode (1 out of 2 trains),
2. Containment Fan Coolers Functional (2 out of 4 fan coolers),
3. Containment Fan Coolers Functional (4 out of 4 fan coolers),
4. Containment Sprays Functional in Recirculation Mode,
5. Containment Isolation Failure (Large Hole Size), and
6. Containment Isolation Failure (Small Hole Size).

To have successful containment heat removal, it is necessary to have two spray trains, one spray train and 2 fan coolers, or 4 fan coolers functional. To properly account for the logic, three nodes were needed in the CSET.

There are 18 possible outcomes for a CSET for each core damage bin, and they are listed in Table 4-17 (page 4-74) of the submittal. In addition, there are 17 possible core damage bins. Hence 306 PDSs are possible. However, to manage this process, the PDSs were truncated by using a cut-off frequency of 10^{-7} per reactor year. After the truncation, a total of 17 PDSs were found to result. The frequency of each PDS, and the representative accident sequence for each PDS is listed in Table 4-18 of the submittal.

The top contributor to core damage frequency is the damage state 1P, a station blackout sequence which progresses into a RCP seal LOCA, and it contributes to 33% of the CDF. Two small LOCAs are the next leading contributors to the CDF. A loss of feedwater transient and an ATWS sequence are the fourth and fifth leading contributors. SGTRs and interfacing system LOCAs contribute to approximately 3.2% of the total CDF.

2.2.2 Containment Event Tree Development

Probabilistic quantification of severe accident progression is performed using an event tree/fault tree methodology, where fault trees are used to quantify each CET node (called "Top Event" in the submittal). Quantification of the events that comprise the CET, and the CET quantification results are discussed in Section 4.6 of the submittal. The CET is concise and contains the following fifteen top events:

(1) In-vessel recovery successful,
(2) Hydrogen burn after in-vessel recovery,
(3) Containment not bypassed,
(4) Successful containment isolation,
(5) Containment leakage rate is small,
(6) Early containment overpressure is prevented,
(7) Containment liner intact,
(8) Ex-vessel cooling successful,
(9) Late hydrogen burn mitigated,
(10) No late containment failure (steam overpressure),
(11) No late containment failure (non-condensible gas),
(12) No basemat failure,
(13) Late revaporization does not occur,
(14) Radionuclide scrubbed by overlying pool, and
(15) Radionuclide scrubbing by containment sprays.

The same CET is quantified for all PDSs.

In-Vessel Recovery

This top event addresses the potential for arresting the core damage progression prior to vessel breach. After core damage, it is assumed that the presence or recovery of AC power is a necessary condition for in-vessel recovery. Two modes of possible depressurization are considered for high pressure sequences, namely, operator action to depressurize and initiate low pressure injection, and induced failure of hot leg or surge line. Even with the restoration of core cooling, the possibility of a non-coolable debris geometry is considered in the fault trees. The negative consequences of restoration of core cooling are considered, namely, in-vessel steam explosions, increased hydrogen generation and the resulting possibility of hydrogen combustion. Qualitatively, the treatment of in-vessel recovery is good, and the level of detail is comparable to NUREG-1150 analyses.

The quantification of the basic events for this fault tree is provided in Table 4-25 (page 4-170) of the submittal. The probabilistic assignments are not supported by the documentation and often appear to be qualitative or unsubstantiated. Table 3 provides a comparison of the values assigned to the basic events between the Surry NUREG-1150 analyses, and those provided in the IPE submittal, for the in-vessel recovery fault tree. Although NUREG-1150 used a large

Table 3 Comparison of Basic Event Probabilities Between the IPE Submittal and NUREG-1150 for In-Vessel Recovery

Event name	Description	Shearon Harris IPE	Surry, NUREG-1150
HOTLFAILS	Hot Leg Fails Due to Thermal Stresses, High Pressure	0.9	0.72 (RCS pressure of 2500 psia)
HOTFAILSX	Hot Leg Fails Due to Thermal Stresses, Intermediate Pressure	0.1	0.03 (RCS pressure less than 2500 psia)
INGEOMOP	Failure of cooling due to non-coolable debris geometry, depressurized by operator action	0.5	0.5 for seal LOCA sequence; 0.1 to 0.3 for station blackout sequences
INGEOMME	Failure of cooling due to non-coolable debris geometry, thermal failure of RCS boundary	0.1	See above
OPERIV	Operator fails to depressurize using PORVs	0.1	0.1 - 0
IVSTEXP	IVSE occurs due to reflood	0.5	Not treated
PSTVMF	RPV fails due to IVSE	0.01	0.008 - 0.0008
BURNGO1, BURNGO2	Hydrogen burn occurs due to reflood	0.001 to 0.1*	Not treated

* Depends on steam concentration in the containment

event tree methodology, a gross comparison of the split fractions (used in the NUREG-1150 event tree top event questions) and the values used for the basic events in the IPE fault tree can be made. Since, the discussion provided in the submittal is mostly qualitative, it is not easy to understand the assigned split fractions. Similarly, no clear justification is provided for the assignment of quantitative probabilities for the lack of coolability due to debris geometry, but the values seem to be comparable to those used for NUREG-1150 analyses. In addition, two deleterious effects of reflood, namely, steam explosions and hydrogen generation and combustion, are quantified. A number of questions were posed to the licensee on the quantification of basic event probabilities. The licensee provided more details of the quantification, and it appears that the licensee has a qualitative understanding of the phenomena.

Hydrogen Burn after In-Vessel Recovery

The treatment of reflood-induced hydrogen generation, combustion and containment failure in the second node of the event tree is detailed. Four factors were considered in determining the potential for hydrogen burn-induced failure of the containment. The first factor is the hydrogen source term. The upper bound of hydrogen generation was set to the mass of hydrogen generated due to oxidation of 75% of the core inventory of zirconium. The results calculated using the MAAP code and other results reported in the literature were used to generate an uncertainty distribution for the hydrogen source term. The second factor was the fraction of

steam in the containment, and this was obtained from the MAAP calculations for each PDS. The third factor was the availability of ignition source. The fourth factor was the containment pressure. The pressure due to hydrogen burns is compared to the containment capacity, and the failure probability is calculated. Typical values of conditional probability of containment failure (with in-vessel recovery) vary from 1.2×10^{-3} for loss of offsite power sequences (i.e. damage state 1P) to 0.227 for damage state 2A, where all containment safeguards are available. The large value of containment failure after in-vessel recovery for damage state 2A is due to the large fraction of in-vessel oxidation (75%) assumed for reflood situation. The treatment of in-vessel hydrogen generation, combustion and containment failure in the IPE submittal is very conservative, and leads to a high probability of containment failure after in-vessel recovery.

Isolation, Bypass and Containment Leakage

Nodes 3 and 4 address the possibility of containment bypass, and the probability of failure to isolate containment, respectively. Node 5 is a summary question that determines the size of the pre-existing leakage from questions 3 and 4.

The treatment of bypass sequences in Node 3 includes both ISLOCA and SGTRs. Both, pre-existing tube rupture (small break LOCA) and thermally-induced steam generator tube rupture are treated in the supporting fault tree. The Emergency Operating Procedures (EOPs) in Shearon Harris plant require the operator to restart the RCP after severe accident initiation in order to provide core cooling. Restart of the RCP leads to loop seal clearance, and establishment of full-loop natural circulation. Heatup of U-tubes by natural circulation can lead to thermally-induced steam generator tube rupture. Secondary side depressurization can increase the potential for induced failure due to increased pressure difference. The submittal assigns a probability of 0.5 for induced steam generator tube failure, if the RCPs are restarted and loop seals cleared during the course of high pressure accidents without secondary side cooling. It appears that the quantification of this probability is based on MAAP calculations. An additional point that is to be noted here is that for cycling steam generator SRVs, the submittal assigns a probability of 0.1 for the failure of SRVs in a stuck open position. For high pressure transients without decay heat removal, where the operators have restarted the RCPs thus leading to loop seal clearing, thermally-induced steam generator tube ruptures are expected to occur. The principal contributors to radiological releases in the Shearon Harris IPE submittal are the thermally-induced SGTR sequences.

Early Containment Overpressure is Prevented

Node 6 is a risk-dominant question that treats early containment overpressurization failure. The effect of the following four energetic events on containment pressure was assessed:

- Ex-vessel steam explosions
- Hydrogen deflagration
- Hydrogen detonation
- DCH

The baseline containment pressure after vessel breach is implicitly included, and obtained from MAAP calculations. Ex-vessel steam explosions are considered to be possible for both high and low pressure cases. The submittal assigns a probability of 0.25 for ex-vessel steam explosion to occur. In contrast, NUREG-1150 assigned a value of 0.5 and zero for low and high pressure scenarios, respectively. The IPE submittal considers containment failure by EVSEs highly improbable, and assigns a probability of 0.001 for containment failure. The reason given in the submittal is that there is no direct pathway from the cavity to the containment wall in the Shearon Harris plant. Failure of the primary shield walls (and subsequent collapse of the RPV and associated RCS piping) is considered to be extremely unlikely. No supporting calculations are performed to support the probabilistic quantification.

Hydrogen deflagration and detonation are treated as possible threats to containment integrity. The best estimate in-vessel hydrogen generation is obtained from the MAAP calculations, with the core blockage model deactivated. The integral mass of hydrogen generated during the course of several severe accidents was then compared with that calculated using other computer codes such as MELPROG and MELCOR. In general, it was found that the in-vessel generation of hydrogen predicted by different codes were comparable, for all accident sequences except the S2 LOCA. For most sequences where CHR was not functional, the containment was found to be steam inert. However, for PDSs with containment safeguards functional (3A, 13A, etc.), hydrogen deflagration was possible, and the resulting containment pressure was calculated. A small conditional probability of 0.001 was assigned for these sequences, for the occurrence of detonations. The basis for this value is not given, except that it is stated that detonations are highly improbable. The probability of containment failure due to hydrogen detonations is assigned a value of 0.25. However, the probability of containment failure due to hydrogen detonations at or around vessel breach is quite low, compared to other failure modes.

Direct containment heating is treated both deterministically and probabilistically in the submittal. In the CET, containment failure due to DCH is assumed to be dependent on RCS pressure. For pressures below 200 psia, HPME and DCH are considered to be impossible. For intermediate pressures (between 200 and 600 psia), containment failure due to DCH is assigned a conditional probability of 0.001. This value was calculated based on a review of MAAP simulations, and some hand calculations. For high pressure scenarios, two sub-cases were considered. For sequences that have a high baseline containment pressure (CHR not functional), a probability of 0.01 was assigned for containment failure. For sequences that have a low baseline pressure, a probability of 0.001 was assigned. Although the submittal refers to calculations performed to obtain these values, these calculations are not documented in the submittal. It is stated in the IPE that calculations have been performed and probabilistic distributions for containment loads were obtained and convoluted with the containment fragility uncertainty distributions to arrive at the conditional probabilities of containment failure due to DCH.

The DCH phenomena is a very complex process, and the magnitude of the DCH-induced containment loads depends on a number of parameters such as the mass of debris in the lower head of the RPV at the time of vessel breach, metallic and oxidic fraction of the debris, mode of vessel breach, fraction of material dispersed and entrained as fine droplets, cavity geometry,

and the mass of co-dispersed cavity water. The design of the cavity and the lower compartment must be carefully considered in determining the fraction of the ejected debris that can be dispersed to the lower compartment. The licensee has performed calculations to assess the HPME-induced DCH loads and the associated conditional containment failure probability.

In addition, a large fraction of high pressure sequences are also assumed to be depressurized, either by natural circulation-induced RCS failure, or by operator action. As a result of these assumptions, the probability of early overpressure containment failure (for which DCH is a significant contributor) is calculated to be extremely small in the submittal. In fact, the overall conditional probability of early overpressure failure is calculated to be 2.5×10^{-3}. The two principal sequences that contribute to early containment failure are small LOCAs with functional failure of ECCS injection and recirculation, respectively. Even these sequences have containment failure associated with hydrogen burns. Sequences such as station blackout scenarios that have been shown to contribute to early containment failure in other PRAs, are shown to result in (almost) zero probability of early containment failure.

Containment Liner Intact

Based on walkdowns and inspection of containment drawings, a Shearon Harris-specific failure mode was identified as a part of the IPE process. In a high pressure sequence, melt ejection from the vessel, sweepout of ejected debris from the cavity into the lower compartment, and the subsequent escape of debris from the lower compartment to the annular compartment through ports in the shield wall, can cause direct contact of debris with the containment liner. However, it must be noted that, in contrast to the BWR Mark-I liner melt-through failure mode, liner failure in the Shearon Harris containment will only lead to leakage of the containment, and no containment rupture is expected to occur. High pressure melt ejection is needed to transport the debris to the annular compartment. If water is present in the annular compartment, it will prevent heatup of the steel liner. Failure of the liner is treated by this node, and a simple model was developed to predict liner failure. A detailed fault tree was constructed for this node. Various split fractions ranging from 0.0001 to 0.1 were assigned to the probability of liner failure by contact with debris, in different scenarios. The probability of liner failure is the highest for the high pressure scenario where the RWST water is not injected in the containment.

Ex-Vessel Cooling Successful

Ex-vessel cooling of debris by RWST water is treated by this node. Four factors are considered to play a role in ex-vessel coolability of debris, namely, status of RWST, RCS pressure at vessel breach, ex-vessel steam explosions (that are assumed to disperse the debris), and potential for forming deep debris pools that cannot be quenched. Deep debris beds can be formed when gravity pours of debris from a breached RPV causes debris to accumulate on the cavity floor. The large floor area (588 ft^2) of the cavity is assumed to aid debris coolability, if conditions favor dispersion of debris. It should be noted that if the entire core inventory of UO_2 and zirconium accumulates on the cavity floor, the resulting debris pool will have a depth of 22 cm. The submittal assumes that the debris will be coolable for a large fraction of sequences.

Late Hydrogen Burn Mitigated

This node addresses the potential for hydrogen burn after vessel failure. Late hydrogen burns can occur as a result of recovery of containment cooling or due to additional hydrogen generation by MCCI. The fault tree shown in Figure 4-32 of the submittal shows that late burn is dependent on a number of factors such as the availability of sufficient hydrogen, recovery of CHR, and upon earlier burns. It should be pointed out that late hydrogen burn-induced containment failure for damage state 1P (loss of offsite power with RCP seal LOCA, no ESF) is a significant contributor to late containment failure in the Shearon Harris plant.

No Late Containment Failure and No Very Late Containment Failure

The two subsequent nodes in the event tree treat containment failure by steaming and containment failure by steaming and non-condensible gas generation, respectively. The former mode of containment failure occurs when a water pool overlies the debris, and the containment is pressurized by steam generated from the overlying pool. This mode of failure is a result of the treatment of water pool-debris-concrete heat transfer in the MAAP code, where the heat transfer is preferential to the overlying water pool. However, AC power can be recovered for a number of PDSs, and the recovery of CHR leads to condensation of steam. The entire inventory of RWST dries out, and the subsequent attack of concrete by core debris leads to non-condensible gas generation, and very late failure of the containment. The very late failure of the containment by non-condensible gas generation is a significant mode of containment failure in the Shearon Harris IPE submittal.

Late Revaporization Does Not Occur

This CET node addresses the issue of whether radionuclides deposited in the RCS piping are released late in the sequence, particularly after containment failure, and thus lead to a higher source term. In a sequence such as a large break LOCA, there is no significant deposition of radionuclides in the RCS, and most of the radionuclides are deposited on the containment structures. For such sequences, no credit is given for late revaporization. For station blackout sequences, late revaporization can lead to large releases, and is an important contributor to the source term.

Radionuclide Scrubbed by Overlying Pool and Radionuclide Scrubbing by Sprays

The two final events treat the possibility of source term reduction due to an overlying water pool and by containment sprays. These two nodes are used for proper binning of the release categories.

In summary, the CET and the fault tree analyses are transparent and treat all the phenomena of interest to severe accident progress in a PWR with a large, dry containment. However, a number of assumptions are made in the quantification, and in some cases, values are assigned to the basic events without justification. A number of questions were posed to the licensee

regarding the details of quantification of the basic events, and their justification. The responses indicate that the licensee has performed more detailed analyses (which are not documented in the submittal), and has gained a qualitative understanding of the severe accident phenomena. The results of the CET analyses are different from the results reported for other PRA studies, and they are discussed further in Section 2.1.3.2.

2.2.3 Containment Failure Modes and Timing

The Shearon Harris IPE submittal makes use of plant-specific calculations performed by consultants (EQE) to determine the ultimate pressure capacity of the containment. The Shearon Harris containment building is a steel-lined concrete shell in the form of a vertical cylinder with a hemispherical dome and a flat base. The base is a (minimum of) 12 ft thick reinforced concrete slab. The containment is lined with a welded steel liner plate, the thickness of which varies from 0.25 inch on the basemat floor to 0.5 inch on the dome wall. The liner thickness is 0.375 inch on the cylinder wall. Four failure locations were identified, namely, the basemat (shear), wall-basemat junction (shear), cylinder membrane, and dome membrane. Access openings, large pipe penetrations, small pipe penetrations, and electrical penetrations were also considered, but they were not found to be controlling. Failure capacities were evaluated at three temperatures, namely, 300°F, 500°F, and 800°F. A review of the capacities indicates that the failures were only weakly linked to temperature, with the capacity being marginally reduced at higher temperatures. It was decided that the reduction with increasing temperature was insignificant, and the capacities evaluated at 300°F were used for the analyses.

Table 4-19 (page 4-78) of the submittal lists the median failure capacities at each location, and Figures 4-17 through 4-20 provide the probabilistic distribution of the failure pressures for each location. It can be seen from the table and the figures that failure at the basemat (where the basemat joins the containment wall) is the dominant failure mode. A median failure pressure of 153 psig, and a mean failure pressure of 150 psig are indicated. This value is about 15-25 psig larger than the containment capacities calculated for the Surry and Zion containments, and is more comparable to that calculated for the Palo Verde and Kewaunee submittals.

The effects of elevated temperature upon containment penetrations were evaluated by a plant-specific analysis and documented by the licensee [8]. The licensee did not include the results as a part of the submittal, but provided a summary as a part of the response to NRC questions. The analyses showed that leakage was possible before containment failure, but the leakage is expected to be small. The licensee stated that electrical penetration failure is not expected to occur since the MAAP calculations showed that the containment temperature was less than 500°F in the severe accident analyses. The containment hatches do not make use of elastomer seals, and hence the licensee stated that failure of elastomer seals under elevated temperature conditions is not an issue in the Shearon Harris plant.

2.2.4 Containment Isolation Failure

A detailed analysis of the containment isolation system in the Shearon Harris plant is provided in Section 3.2.17 (page 3-178) of the submittal. Mechanical and electrical penetrations were evaluated to determine the probability of failure. Owing to the existence of a large number of penetrations, a screening process was used to determine which penetrations have to be analyzed in detail. Electrical penetration assemblies are assumed not to fail, based on the passive design of the assemblies, and the protection scheme for overcurrent conditions. A total of 110 mechanical penetrations were identified, but the screening process led to the identification of 9 penetrations that required detailed analyses. The bases for the exclusion of the remaining penetrations are provided in Table 3-21 (page 3-183) of the submittal. The penetrations identified for further analyses include the following:

- Seal water return and excess letdown line,
- Two containment spray injection, and two pump suction lines (4 penetrations),
- Two RHR pump suction lines,
- Containment sump pump discharge, and
- Personnel air lock.

Fault trees were developed to quantify the probability of isolation failure for these lines. An example fault tree (used for evaluating the failure of one of the penetrations to isolate containment) was provided as a part of the licensee response to the NRC questions [8]. The calculated frequency of the failure of the containment isolation system is $2x10^{-7}$ per reactor year, and is similar to that estimated in other IPE submittals (i.e., $8.9x10^{-6}$ per reactor year for Zion, and $4.1x10^{-7}$ per reactor year for Turkey Point).

Containment bypass was also analyzed as a part of the IPE. All systems interfacing with the RCS were identified and screened to assess the potential for ISLOCA. Three paths for ISLOCA were judged to be potentially significant, and they are:

- *RHR suction line* - Rupture is similar to a large LOCA, and a frequency of $2.7x10^{-7}$ per reactor year is calculated for this bypass sequence.

- *Cold leg safety injection line* - Leakage of the check valve between the high pressure RCS and the low head safety injection system can lead to an ISLOCA, and a frequency of $1.9x10^{-7}$ per reactor year is calculated for this event.

- *Hot leg safety injection line* - The ISLOCA in the hot leg safety injection line can occur in a manner similar to that for the cold leg safety injection, but the number of check valves and injection lines are different for the hot leg injection. A frequency of $4.8x10^{-8}$ per reactor year is calculated for this sequence.

In summary, two events representing ISLOCAs were identified for the Shearon Harris plant, these include; a large break ISLOCA (in the RHR suction line), with an initiating frequency of

2.7×10^{-7} per reactor year, and a medium break ISLOCA (in the safety injection lines), with a combined initiating event frequency of 2.3×10^{-7} per reactor year. The calculated frequency for the bypass sequences is similar to those calculated for the Zion NUREG-1150 analyses, and several other IPEs reviewed to date.

2.2.5 System/Human Response

Two operator actions were found to be modelled in the Shearon Harris IPE submittal containment event tree, and they are:

- Operators fail to depressurize RCS, and
- Operators preclude hydrogen combustion following recovery of the CHR systems (by inhibiting the CHR systems thus maintaining a high steam concentration within the containment atmosphere) in the late phase of accidents.

The first operator action is assigned a probability of 0.1, since operator action to depressurize the RCS is considered highly likely. The details of the quantification of this basic event are not given; however, it should be noted that a similar value is also assigned in the NUREG-1150/ Surry analyses. The second operator action considers the possibility that the operator will not use the spray pumps and fan coolers, after recovery of these containment heat removal systems late in the course of the accident. The submittal considers this action unlikely, and assigns a value of 0.001 for this basic event. This has a negative impact on the accident progression, since the failure of the operator to control the steam concentration can lead to late hydrogen burns that can fail the containment.

2.2.6 Radionuclide Release Categories and Characterization

The results of the CET analyses lead to an extensive number of end-states, which are classified into a manageable number of release categories, characterized by similarities in accident progression and source term characteristics. The main characteristics of the CET end-states considered when developing these release categories in the submittal were:

- Release energy,
- Containment isolation failure size,
- Time of failure, and
- Isotopic composition

However, after the study was completed, it was determined that the energy of release was not as significant in the framework of the IPE, and therefore all early overpressure failures and large isolation failures were grouped into one release bin. Similarly, all late failures including both gradual overpressurization failure and hydrogen burn-induced failures were grouped together.

The second bin characteristic defines whether or not the containment was isolated from the initiation of the accident, and three alternatives were chosen for this parameter: large, small, and intact.

The timing of release impacts the source term and the protective action warning time. Early release is defined to occur prior to, at, or around the time of vessel breach. Late release is defined to occur after RPV failure until 48 hours following accident initiation. Very late release refers to releases that occur a few days after accident initiation.

Based on the isotopic composition of the source term, the releases were classified into the following:

- Releases corresponding to an intact containment,
- Releases in which MCCI occurs,
- Releases in which revaporization occurs, and
- Releases which are scrubbed.

Although 72 combinations are possible, for purposes of simplification, the CET end-states were grouped into 15 bins or release categories. The binning rationale and the accident sequences that were used to characterize each release category are discussed in Table 4-28 (page 4-186) of the submittal. Table 4-31 (page 4-191) of the submittal provides the details of the source term results for all the release categories.

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 BWR-3 or PWR-4 release categories of WASH-1400." should be reported by the IPEs. The IPE submittal appears to fulfill this request, and it is also possible to obtain these sequences from the information provided in the submittal.

Functional sequences that contribute to more than 10^{-6} per reactor year are listed in pages 3-243 and 3-244 of the submittal. There are six such sequences. A transient-induced seal LOCA is the sequence with the largest contribution to core damage frequency. Two small break LOCAs with failure of injection and recirculation, respectively, are the next two dominant sequences. A sequence involving loss of all decay heat removal is the next leading contributor, followed by an ATWS sequence, and an internal flooding sequence that leads to RCP seal LOCA. However, only one of these sequences leads to releases that exceeds the PWR-4 releases of WASH-1400. That sequence is the loss of decay heat removal sequence, which progresses into an induced SGTR sequence. This sequence has large releases of volatiles. In addition, Generic Letter 88-20 also requests the reporting of all bypass sequences with a CDF of more than 10^{-7} per reactor year. As already described in Section 2.1.2.4, there are two ISLOCA sequences with CDF greater than 10^{-7} per reactor year. Releases were not calculated for these sequences, and the source term, for the bypass sequences were represented by the RC-5 category.

Release categories 4 and 5 have the largest frequency of releases, and they represent steam generator tube rupture sequences. A large fraction of the RC-5 releases is due to a thermally-induced SGTR for a transient which originated as a decay heat removal failure sequence. The difference between RC-4 and RC-5 is that RC-4 involves cycling SRVs and the releases are scrubbed by the secondary side water inventory. RC-5 represents a SGTR with a stuck-open SRV, and the releases are unscrubbed. Together, these two release categories represent more than 60% of the total release frequency. A third SGTR release, namely, RC-4C, which represents unscrubbed releases from a cycling SRV, represents the third largest source term in magnitude. SGTRs are the leading contributors to source term releases in the Shearon Harris IPE submittal.

A comparison of release magnitudes for the steam generator tube rupture sequences estimated by the IPE submittal to those reported in the literature for other plants based on STCP [6] and MELCOR [7], is shown in Tables 4 and 5. Table 4 compares the releases for a SGTR with a stuck open SRV. Table 5 represents the releases for a cycling SRV. Both releases are not scrubbed, and results for scrubbed releases are not available in the literature. Scrubbed releases are not expected to be large. These scrubbed releases are influenced by various other factors such as secondary side inventory, auxiliary feedwater flow, and decontamination factors used in the analyses, and hence, they are not compared here. The results indicate that the iodine and cesium releases reported in the submittal (for the stuck-open SRV accident) are larger than the STCP results or the MELCOR results. Tellurium releases are two orders of magnitude smaller for the Shearon Harris calculations. The submittal also reports Strontium, Barium, Cerium and Lanthanum group releases, which are one order of magnitude higher than the BCD results. The differences in the releases for the volatile groups such as the iodine and cesium groups are probably due to the high degree of revaporization predicted by the MAAP code. The differences in the releases of tellurium, strontium and barium are probably due to the treatment of MCCI in the MAAP code, which models a preferential heat transfer to a pool of overlying water.

Table 5 shows a comparison for the SGTR with a cycling SRV. The volatile releases predicted by the MAAP code are a factor of 2 lower than those predicted by STCP. The release magnitude of the nonvolatile species are considerably lower.

No attempt was made to compare releases for other sequences such as the small break LOCA and the station blackout sequence. The submittal shows that the releases for these sequences occur by containment failure due to slow overpressurization occurring days after accident initiation. Such releases are not expected to be large, owing to considerable retention of fission products within the containment.

Figure 1 shows a plot of the frequency of exceedance of iodine release as a function of the fraction of inventory released to the environment. The results for the Surry and Zion plants from the NUREG-1150 analysis are also plotted in the same figure. However, it can be seen that the release profile for the Shearon Harris plant is quite different than that of the other two plants, since it is dominated by steam generator tube rupture sequences.

Table 4 Comparison of Releases for Surry ("H" SGTR Sequence) and Shearon Harris (SGTR Sequence, Release Category 5)

Group	W PWR (MELCOR) [7]	Surry (STCP) [6]	Shearon Harris (MAAP) [1]
I	0.28	0.25	0.59
Cs	0.28	0.25	0.59
Te	0.06	0.08	1E-4
Sr	3E-3	2E-4	1E-4
Ru	1E-4	3E-7	NA+
La	1E-5	2E-8	9E-5
Ce	2E-5	0.0	1E-4
Ba	NA+	3E-3	0.0007

Table 5 Comparison of Releases for Surry ("G" SGTR Sequence) and Shearon Harris (SGTR Sequence, Release Category 4C)

Group	Surry (STCP) [6]	Shearon Harris (MAAP) [1]
I	0.09	0.037
Cs	0.08	0.037
Te	0.04	0.0
Sr	4E-5	8.4E-6
Ru	1E-7	NA+
La	1E-8	1.3E-5
Ce	0	1.3E-5
Ba	5E-4	5.0E-5

+ NA: Not Available

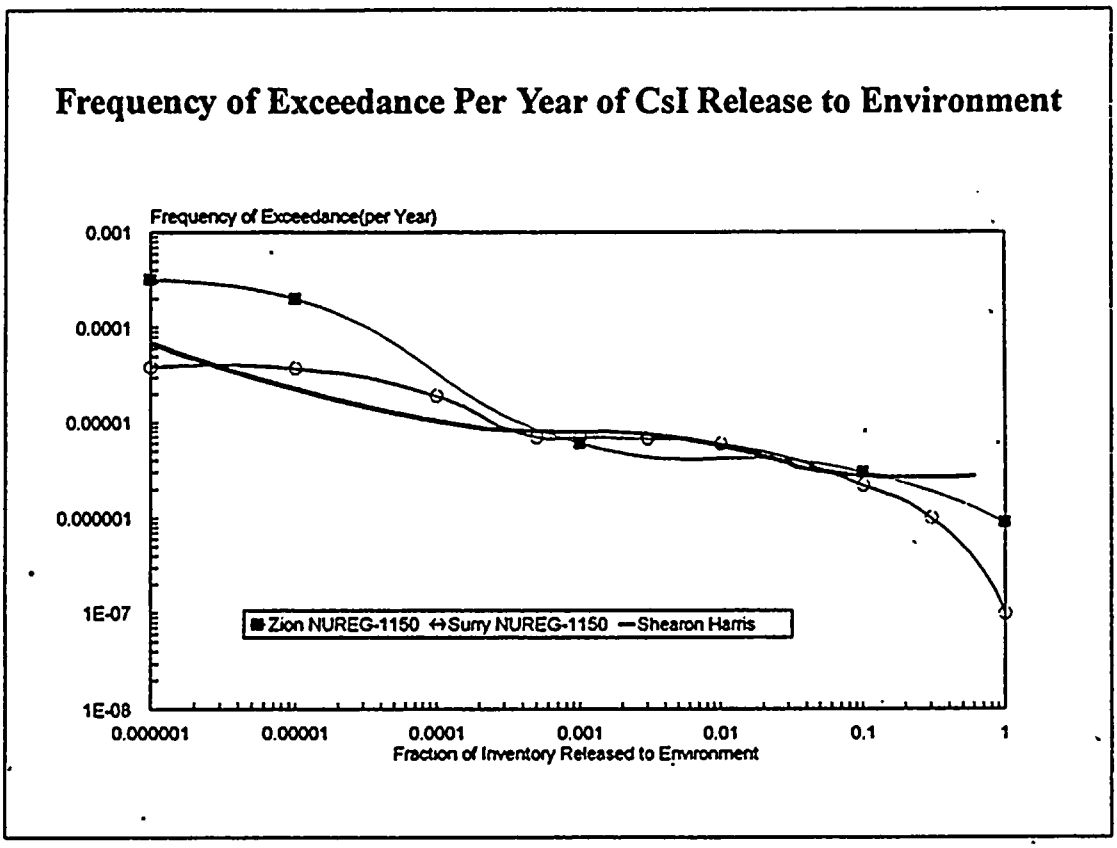

Figure 1 Frequency of Exceedance of Iodine Release

2.3 Quantitative Assessment of Accident Progression and Containment Behavior

2.3.1 Severe Accident Progression

The MAAP 3.0B code (version 19) was used to evaluate the integrated containment response and the severe accident source terms. Although mention is made of the MAAP parameter input file, it is not included in the submittal. The MAAP analyses performed for severe accident analyses and source term quantification are very detailed, and a total of 15 sequences were analyzed. The choice of sequences to be analyzed was based on the CET analyses in order to support the quantification of split fractions for the CET, and to enable the proper quantification of source terms for all important end-states. A listing of all the analyzed accident sequences is provided in Table 4-3 (page 4-22) of the submittal.

The "base case" MAAP calculations were performed with a set of nominal model parameter values which are recommended in Reference [2]. The following assumptions were made in the base case calculations:

1. High pressure sequences were modelled with and without an induced rupture of the hot leg.

2. Vessel failure was assumed to occur 1 minute after debris relocation to the lower plenum.

3. The median containment failure pressure was assumed to be 150 psig, and the failure area was assumed to be 0.5 m^2.

4. The core blockage model in the MAAP code, which leads to lower integral hydrogen generation was deactivated.

5. For high pressure sequences, after vessel breach, 3% of the core debris was assumed to be finely fragmented and dispersed in the containment for participation in DCH.

6. The heat flux from core debris in the cavity to the overlying pool of water was fine-tuned to be 83.7 kW/m^2. This leads to steaming, and preferential heat transfer to the overlying water pool.

7. All fission product tellurium was assumed to bind to unoxidized zirconium (during the in-vessel phase), and therefore can only be released from the core debris during the MCCI phase of the accidents.

Some of these assumptions are not fully supported by experimental evidence available in the literature.

2.3.2 Dominant Contributors to Containment Failure

The containment failure modes and timings for various accident sequences are provided in Section 4.6.3 and summarized in Section 4.8 of the submittal. Table 6 of this review shows a comparison of the conditional probabilities of the various containment failure modes of the Shearon Harris IPE submittal with the Surry and Zion NUREG-1150 results. All comparisons are made for internal initiating events only.

Table 6 Containment Failure as a Percentage of Total CDF: Comparison With Other PRA Studies

Containment Failure Mode	Shearon Harris IPE	Surry NUREG-1150	Zion NUREG-1150
Very Early Failure with In-Vessel Recovery (Prior to Vessel Breach)	3.2	-**	-**
Early Failure	0.25	0.7	0.5
Late Failure	1.0	5.9	24.0
Very Late Failure	3.6	NA+	NA+
Bypass (V)	0.7	7.6	0.2
Bypass (SGTR)	6.5	4.6	0.3
Isolation Failure	0.3	NA++	1.0
Intact	84.5	81.2	73.0
Core Damage Frequency, yr^{-1}	$7x10^{-5}$*	$4.1x10^{-5}$	$6.2x10^{-5}$

* Includes Flooding
** Not Applicable
\+ Included as a Part of Early Containment Failure
\++ Included as a Part of Late Containment Failure

The Shearon Harris core damage frequency for internal events is slightly larger than that calculated by NUREG-1150 for Surry and Zion [4,5]. The conditional probability of early containment failure (due to overpressurization) in the Shearon Harris plant is 0.25% and is considerably less than that calculated for the Zion and Surry plants. This is primarily due to the treatment of the phenomena that threaten the containment integrity at vessel breach, such as DCH, steam explosions, etc. The RCS is assumed to be depressurized at vessel breach for a large fraction of accident sequences in the Shearon Harris IPE. However, the submittal

calculates a large conditional probability of containment failure before vessel breach (3.2%) with in-vessel core damage arrest. Containment failure is presumed to occur due to additional hydrogen generation due to reflood and subsequent combustion. Late failure as defined in the submittal occurs within 48 hours following accident initiation, and is caused by hydrogen burn late in the course of the accident. Very late failures are caused by overpressurization of the containment by steam and non-condensible gas generation. The total conditional probability of 4.6% calculated for late and very late failure for the IPE submittal is roughly comparable to the conditional probability of late containment failure calculated for the Surry plant. The largest contribution to containment failure in the IPE submittal is from the SGTR sequences, and they are of two different types. Roughly half of the SGTR sequences have tube ruptures at the beginning of the accident sequence, and they are the traditional unisolated SGTR sequences. The other half of the SGTR failure mode is contributed by damage state 2A, and it involves a thermally-induced SGTR. This occurs due to a plant-specific procedure for this class of accident sequences that requires RCP startup during the course of the accident. Hot gases are pumped by the RCP to the steam generator tubes, and lead to tube failure. This is a plant and procedure-specific mode of containment failure identified by the submittal.

2.3.3 Characterization of Containment Performance

To characterize the containment performance under severe accident conditions, the licensee considered the uncertainty in important phenomenological events during severe accident progression. The key areas of uncertainty in the response of the SHNPP primary system and containment to a severe accident are stated to be the following [1]:

1. Magnitude of hydrogen source term and threat to containment integrity posed by hydrogen deflagrations and detonation,

2. Vessel breach (and prevention of vessel failure by cavity flooding),

3. Temperature-induced rupture of the hot leg, steam generators, and surge line,

4. Size and timing of vessel breach, including the impact on containment loads,

5. HPME, DCH, and direct attack of containment liner by dispersed debris,

6. Coolability of core debris in the containment, and

7. Magnitude of fission product revaporization from the RCS.

The primary tool used for sensitivity analyses was the MAAP code. Hand calculations were performed for phenomena not modelled in MAAP. In addition, existing literature was reviewed to determine the uncertainty in phenomena. The results of these analyses are stated to have been used to assign the basic event probabilities in the phenomenological fault trees.

2.3.4 Impact on Equipment Behavior

The impact of the accident progression on equipment performance after core damage was not considered as a part of the CET analyses (or deterministic analyses) in the IPE submittal. Possible equipment failure that can impact severe accident progression include failure of sprays, fan coolers, and failure of ECCS recirculation function due to plugging of the ECCS sump by debris. The lack of treatment of the impact of severe accident progression on equipment behavior is one of the shortcomings of the submittal.

2.3.5 Uncertainty and Sensitivity Analysis

A number of sensitivity studies were performed using the MAAP code, and they include the following:

1. *Hydrogen production and combustion:* Sensitivity calculations were performed to study the uncertainty of the in-vessel hydrogen generation, hydrogen generation and combustion during DCH, and hydrogen generation during MCCI.

2. *Timing and mode of vessel failure:* Sensitivity calculations were performed by varying the RPV failure size and failure time.

3. *Debris entrainment from the cavity and DCH:* Sensitivity calculations were performed by varying the size of vessel failure, and by considering induced RCS ruptures.

4. *Debris coolability on the cavity floor:* Sensitivity calculations were performed by varying the heat transfer from the molten debris to the overlying pool of water from 800 kW/m^2 to 150 kW/m^2.

5. *Revaporization of fission products from the RCS:* Appropriate input parameters were varied so that the settled fission products were forced to revaporize at containment failure.

These sensitivity calculations were performed only for the station blackout and the small break LOCA sequences.

The base case calculations performed for in-vessel hydrogen generation for various accident sequences were found to be comparable to the results from the NRC codes such as MELPROG and MELCOR. Detailed hydrogen burn and containment pressurization calculations were performed outside of the MAAP sensitivity calculations. The parametric range of the in-vessel hydrogen source term was augmented by the distributions provided by the experts for the NUREG-1150 analyses for the Surry plant. In general, it was found that hydrogen combustion-induced containment failure was possible only for two accident sequences.

High pressure melt ejection was analyzed by a number of sensitivity calculations performed using the MAAP code. The table provided in page 4-46 of the submittal gives an overall summary of the resuits, considering the effect of the fraction of the core inventory fragmented, the time constant for dispersal, and the effect of forcing hydrogen to burn continuously. The maximum containment pressure calculated was 110 psia, by assuming that the entire core inventory participated in DCH, and all the hydrogen generated was burnt. For all other cases, the calculated peak containment pressure was lower. Based on the MAAP sensitivity studies, and the fact that the maximum calculated containment pressure was considerably lower than the containment median capacity, the submittal concludes that direct containment heating is not a threat to containment integrity. However, the uncertainty distribution for DCH-induced pressurization loads was not developed and convoluted with the uncertainty distribution for the containment structural fragility, and thus the conditional probability of failure was not calculated.

The effect of the reduced heat transfer to the overlying pool of water, studied to investigate core coolability in the cavity, was to increase the erosion depth and generation of hydrogen and other non-condensible gases. These results do not appear to have been used in the CET analyses.

In addition to the MAAP sensitivity analyses, the submittal makes use of hand calculations to study liner failure by contact with molten debris. Details of this simple model are provide on page 4-48 of the submittal. Owing to the large floor area of the cavity, the height of the debris pool is not expected to be more than 10 cm. The liner temperature obtained by using a two dimensional heat conduction model is plotted as a function of debris pool height in Figure 4-15 (page 4-50) of the submittal. It is concluded that, for sequences where the RWST is injected into the containment, liner failure is unlikely. For low pressure sequences, the core debris cannot reach the annular compartment of the containment. Hence, debris contact with the liner and subsequent liner failure is possible only for high pressure sequences where the RWST is not injected. Even in such cases, it is not feasible that the mass of the debris that reaches the annular compartment and the liner will be large enough to heat up the liner.

The primary tool used for sensitivity and uncertainty analyses was the MAAP code. Hand calculations were performed for phenomena not modelled in MAAP. In addition, existing literature was reviewed to determine the uncertainty in phenomena. The results of these analyses are stated to have been used to assign the basic event probabilities in the phenomenological fault trees.

2.4 Reducing the Probability of Core Damage and Fission Product Releases

2.4.1 Definition of Vulnerability

The submittal does not define "vulnerability", particularly as related to containment analyses. However, it should be noted that a few initiators, particularly the loss of offsite power and the small break LOCA, dominate the CDF profile. In addition, it should also be noted that the EOP requiring the operators to restart the RCPs to provide additional cooling during the course of severe accidents was found to have important consequences. In the submittal, this operator

action is shown to lead to induced SGTR and thereby large radiological releases. In response to an NRC question on the induced SGTR (due to reactor coolant pump restart), the licensee stated that this failure mode of steam generator tubes is not unique to the Shearon Harris plant, and that the issue is being addressed by the Westinghouse Owners Group.

2.4.2 Plant Modifications

A multi-disciplinary utility team was formed to review the IPE results and suggest plant modifications and improvements. The team recommended changes to plant procedures to allow for manual operation of breakers in transferring to offsite AC power, and recommended investigations for further improving the reliability of the DC system. However, this team made no recommendations for any improvements based on the back-end analyses, and hence no containment-related improvements were considered necessary.

2.5 Responses to CPI Program Recommendations

One of the recommendations of the CPI program pertaining to PWRs with large dry containments was that the utility should evaluate the IPE results for containment and equipment vulnerabilities to hydrogen combustion (local and global), and point out any need for procedural and/or hardware improvements. The submittal documentation does not explicitly discuss the recommendations of the CPI program. However, in response to the NRC questions, the licensee discussed the recommendations of the CPI program, and their treatment of the CPI program recommendations [8]. The licensee has studied the impact of local and global hydrogen combustion upon containment and equipment performance at Shearon Harris. However, it was stated that no recommendations could be identified for improving containment performance relevant to global hydrogen combustion at SHNPP. With regards to the localized hydrogen combustion, a quantitative assessment of hydrogen release points (pressurizer relief tank, hot and cold legs, etc.) by the licensee failed to identify any enclosed spaces that would allow for hydrogen buildup and combustion that can endanger equipment. In addition, it was noted that most of the safety grade equipment is located outside the containment, and is not impacted by adverse environment inside the containment. The only important systems located inside the containment are the containment sprays and fan coolers. The licensee stated that the spray nozzles are reasonably insensitive to hydrogen combustion, but the fan coolers have the potential to be impacted by a local hydrogen burn. The likelihood of the failure of fans was quantitatively analyzed, and found to be rather insignificant. Two factors helped to reduce the significance of local hydrogen combustion. First, many accident sequences (that lead to core damage) involve failure of support systems, and fan coolers were not functional during accident progression. For sequences that had fan coolers functional, but unable to operate (i.e, due to unavailability of AC power), recovery of AC power caused both fan coolers and sprays to operate. With either system being sufficient for containment heat removal, the fan coolers can fail without adversely impacting accident progression. Hence, the overall IPE results for containment failure and the radionuclide release frequency were not found to be impacted by the failure of fan coolers due to local hydrogen combustion. In addition, the licensee noted that a review of the results from severe accident analyses did not identify any volumes inside the

containment where a sufficient buildup of hydrogen could lead to DDT. However, the licensee stated that a containment walkdown has not been performed to identify passages conducive to the occurrence of DDT. In summary, all the CPI recommendations are addressed by the licensee.

3. CONTRACTOR OBSERVATIONS AND CONCLUSIONS

The ERI assessment of the Step 1 review is that the Shearon Harris IPE submittal contains substantial *Back-End* information regarding the severe accident vulnerability issues for the SHNPP. The IPE uses a small event tree/large fault tree methodology to perform the probabilistic segment of the back-end analyses, and uses the MAAP code to perform the deterministic accident analyses. However, the submittal uses the results from the MAAP simulations and open literature to exclude most of the early challenges to the containment integrity.

The important points of the technical evaluation of the Shearon Harris IPE back-end analysis are summarized below:

- The Back-End portion of the IPE supplies a substantial amount of information with regards to the subject areas identified in Generic Letter 88-20.

- The Shearon Harris IPE provides an evaluation of all phenomena of importance to severe accident progression in accordance with Appendix I of the Generic Letter.

- Phenomena that lead to early containment failure (i.e., direct heating, hydrogen combustion) have been treated in considerable detail. The licensee has attempted to quantify the uncertainty in phenomena such as Direct Containment Heating (DCH) and hydrogen combustion. The submittal has also identified a plant-specific containment failure mode that involves direct liner attack by high pressure-induced dispersal of core debris, and quantified the probability of liner failure due to direct contact by core debris. The results of the containment failure analysis show that early containment failure for the Shearon Harris plant is dominated by hydrogen combustion.

- The licensee has evaluated the positive and negative consequences of operator actions to add water and cool a degraded core. Additional hydrogen generation by core reflood, possibility of in-vessel steam explosions, and thermally-induced steam generator tube rupture by RCP restart, have all been treated in the submittal.

- The submittal has identified one specific procedural action that exacerbates the outcome of certain severe accident sequences. A procedure that requires the operator to restart the RCP to provide core cooling was found to lead to thermally-induced steam generator tube rupture. The submittal does not identify any corrective actions for this procedure. In response to an NRC question on the induced SGTR (due to reactor coolant pump restart), the licensee stated that the failure mode of steam generator tubes is not unique to the Shearon Harris plant, and that the issue is being addressed by the Westinghouse Owners Group. It should be noted that this potential failure mode has not been identified by other IPEs reviewed by ERI, for other Westinghouse PWRs.

- The licensee has addressed the recommendations of the CPI program, as a response to NRC review team questions.

An utility review team was formed to make use of the IPE results. The team recommended changes to plant procedures to allow for manual operation of breakers in transferring to offsite AC power, and recommended investigations for further improving the reliability of the DC power system. However, this team appears to have made no recommendations for any improvements based on the back-end analyses. Hence, the licensee has no plans for modifications or improvements based on the IPE back-end results.

4. REFERENCES

1. "Shearon Harris Nuclear Station Individual Plant Examination Submittal Report," prepared by Carolina Power and Light Company (January 1993).

2. "Severe Accident Risk: An Assessment of Five U.S. Nuclear Power Plants," NUREG-1150 (1990).

3. Kenton, M. K., and Gabor, J. R., "Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP 3.0B," EPRI report (1988).

4. "Evaluation of Severe Accident Risks: Surry Unit 1", U. S. Nuclear Regulatory Commission, NUREG/CR-4551, Vol. 3, Part 1 (June 1990).

5. "Evaluation of Severe Accident Risks: Zion Unit 1", U. S. Nuclear Regulatory Commission, NUREG/CR-4551, Vol. 7, Part 1 (June 1990).

6. R. S. Denning et al., "Radionuclide Release Calculations for Selected Severe Accident Scenarios, Supplemental Calculations," U. S. Nuclear Regulatory Commission, NUREG-4551, Vol. 6 (August 1990).

7. R. Vijaykumar, M. Khatib-Rahbar, I. K. Madni, E. G. Cazzoli, H. P. Isaac, and U. Schmocker, "Simulation of Severe Reactor Accidents: A Comparison of MELCOR and MAAP Computer Codes", Paper Presented at the ANS Probabilistic Safety Assessment International Topical Meeting, Clearwater Beach, Florida (January 26-29, 1993).

8. "Responses to NRC Questions on the Shearon Harris Nuclear Power Plant Individual Plant Examination Submittal," prepared by Carolina Power and Light Company (1995).

APPENDIX A

IPE EVALUATION AND DATA SUMMARY SHEET

PWR Back-End Facts

Plant Name

Shearon Harris

Containment Type

Large dry containment

Unique Containment Features

Quartz-based aggregate concrete

Large cavity floor area

Unique Vessel Features

Integrated reactor vessel head

Number of Plant Damage States

17

Containment Failure Pressure

Dry 150 psig (median)

Additional Radionuclide Transport and Retention Structures

Auxiliary building structures have not been credited

Conditional Probability That The Containment Is Not Isolated

0.003

Important Insights

Cavity design does not permit debris transport to the upper compartment, and large cavity floor area aids debris coolability

Unique Safety Features

None identified

Implemented Plant improvements

No plant improvements were found necessary

C-Matrix

See Table 4-26 (page 4-174) of the submittal. The C-Matrix includes 6 of the 17 original PDSs.

APPENDIX C

SHEARON HARRIS NUCLEAR PLANT INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(HUMAN RELIABILITY ANALYSIS)