ERI/NRC 94-112

TECHNICAL EVALUATION REPORT OF THE HOPE CREEK NUCLEAR STATION INDIVIDUAL PLANT EXAMINATION BACK-END SUBMITTAL

FINAL REPORT

December 1995

Energy Research, Inc.
P.O. Box 2034
Rockville, Maryland 20847

Prepared for:

SCIENTECH, Inc.
Rockville, Maryland 20852

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 Public Service Electric and Gas Company's (PSE&G) Individual Plant Examination (IPE) submittal of the Hope Creek Generating Station (HCGS). 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 in response to NRC questions.

The back-end portion of the IPE submittal supplies a substantial amount of information with regards to the subject areas identified in Generic Letter 88-20, and NUREG-1335.

E.1 Plant Characterization

The Hope Creek plant is a General Electric Company BWR/4-251 (251 inch diameter vessel) with a Mark I containment. The Hope Creek plant is very similar to the Peach Bottom plant. The rated thermal power is 3293 MWt (1067 MWe). The mean containment failure pressure is 120 psig.

E.2 Licensee's IPE Process

The IPE was a cooperative utility-contractor effort, with most of the work being performed by PSE&G staff. The contractor assistance can be summarized as follows: Halliburton NUS provided training and the primary leadership for the baseline quantification tasks. Science Application International Corporation (SAIC) provided technical direction in areas of event-tree development, special initiating events analysis, human reliability analysis, and cutset editing and analysis. SAIC also performed an HCGS-specific containment bypass analysis, and provided technical expertise and training for the IPE back-end analysis. Gabor, Kenton and Associates were used by PSE&G to review a portion of the MAAP parameter file. ERIN provided the primary leadership and performed evaluations for the plant-specific Interfacing Systems LOCA analysis. ABB Impell performed a containment capacity analysis for the HCGS IPE. The IPE submittal is essentially complete with regard to the recommendations of Generic Letter (GL) 88-20 and NUREG-1335.

A small event tree/large fault tree approach was employed in the front-end analysis, producing a moderate number (81) of core-damage accident sequences having a frequency in excess of 10^{-10} per reactor year (internal cut-off frequency in HCGS IPE). The frequencies of these 81 sequences ranged from the highest of 3.27×10^{-5} per reactor year down to the lowest of 1.4×10^{-10} per reactor year. Of the 81 core-damage sequences produced in the Level I analysis, 17 dominant sequences are screened for evaluation in the Level II analysis, based on the screening criteria of NUREG-1335 (Section 2.1.6). The 17 dominant core-damage sequences, used as initiators for the Level II analysis, together account for more than 95% of the Core Damage Frequency (CDF), and they include all core-damage sequences with a frequency greater than 10^{-7}

per reactor year. The binning criteria for the HCGS IPE were developed based on the criteria applied in the NUREG-1150 analysis of Peach Bottom Unit 2.

The submittal reports a CDF of 4.58×10^{-5} per reactor year. The dominant contributors to core damage are Long-Term Station Black-Out (LT-SBO) sequences (73.8%), followed by transient sequences (14.8%), Loss of Coolant Accidents (LOCAs, 6.7%), special initiators (3.1%), and Anticipated Transient Without Scram (ATWS) sequences (1.6%). ISLOCA and containment bypass sequences are not significant contributors (frequency of 1.7×10^{-9} per reactor year).

Probabilistic quantification of severe accident progression for the probabilistically significant accident bins was performed using Containment Event Trees (CETs). The methodology employed in the Hope Creek IPE submittal involved "linked" event trees, where linking implies that there are common events among the event trees. The CET has seventy-eight nodes, which are further developed using subtrees. The event tree includes information on the following:

1. Vessel depressurization.
2. Injection recovery in-vessel.
3. Vessel failure (core damage arrest in-vessel).
4. Early containment failure.
5. Early release to suppression pool.
6. Operation of drywell sprays.
7. Injection provided to debris following vessel failure.
8. Coolable debris forms ex-vessel.
9. Late containment failure.
10. Late release to suppression pool.
11. Fission product retention.
12. Secondary containment retention.
13. Vent operation.

The CET is evaluated using the EVNTRE code developed for NUREG-1150. In a qualitative sense, the CET methodology is similar to the large event tree method used for the NUREG-1150 back-end analyses, and it appears that the use of subtrees and sub-subtrees is primarily for the purpose of illustration. The branch point probabilities in the subtrees (referred to as basic events) are assigned by the IPE analysts. CET quantification is performed using results from plant-specific MAAP analyses and published results calculated using other codes such as CONTAIN and MELCOR. Results from NUREG-1150 analyses were used to supplement the available results, and to envelop the ranges of phenomenological uncertainty. Results from more up-to-date literature are used for the treatment of uncertainties in drywell shell melt-through. The overall methodology employed in the Hope Creek IPE submittal for CET analysis is well organized. The Hope Creek CET includes all the relevant severe accident phenomena applicable to BWRs with Mark I containments.

The CET end-states are binned into radiological release categories. The binning of release categories was based on the timing of the releases, and the magnitude of iodine and tellurium releases. A source term algorithm was developed to evaluate the magnitude of release fractions for the various release categories. In addition to the base case CET analyses, a number of sensitivity cases were also studied, and some important insights were obtained.

E.3 Back-End Analysis

The conditional probabilities of early and late containment failure calculated by the submittal are 0.56 and 0.18, respectively (see Table E.1). The conditional probabilities of early and late venting are 0.06 and 0.10, respectively. The conditional probability of intact containment is about 0.10. The frequency of early high and early medium releases are 9.42×10^{-6} per reactor year and 6.14×10^{-6} per reactor year, respectively. The conditional probabilities and absolute frequencies of the early-high and early-medium releases are large. Long-term station blackout sequences are the dominant contributors to early containment failure (88%) and late containment failure (74%). The dominant contributor to early releases (79%) and late releases (68%) is the TeEDG sequence (long-term station blackout sequence). This sequence is initiated by loss of offsite power followed by failure of emergency diesel generators, failure of HPCI and RCIC due to battery failure after four hours, and core damage within two hours of battery failure. The largest contributor to early containment failure (by failure mode) is drywell liner melt-through, followed by overpressure. The largest contributor to late containment failure is late overtemperature, followed by sump ablation. The identification of sump ablation as a possible failure mode, together with the higher conditional probabilities of late overtemperature failure, together lead to the calculation of higher conditional probabilities of late containment failure in the IPE submittal.

Table E.1 Containment Failure as a Percentage of Internal Events CDF: Comparison of Hope Creek IPE Results to Peach Bottom NUREG-1150 Results

Containment Failure	Peach Bottom NUREG-1150	Hope Creek IPE
CDF (per year)	4.5×10^{-6}	4.58×10^{-5}
Early Failure	56	62^{+}
Bypass	NA	-
Late Failure	16	28^{++}
Intact	18	10*
No Vessel Breach	10	-

NA - Not Available
* - Includes Both Intact Containment and No Vessel Breach Cases
\+ - Includes the Following Breakup: Structural Failure = 55.7%; Venting = 6.4%
++ - Includes the Following Breakup: Structural Failure = 18%; Venting = 10.4%

The licensee's process for the evaluation of containment failure probabilities and failure modes is consistent with the intent of Generic Letter 88-20, Appendix I. The dominant contributors to containment failure are consistent with the insights obtained from the NUREG-1150 analyses for the Peach Bottom plant. The licensee has considered the failure of the containment isolation system and containment bypass scenarios. A number of sensitivity analyses have also been performed.

E.4 Containment Performance Improvements

Generic Letter 88-20, Supplement Numbers 1 and 3 identified specific Containment Performance Improvements (CPIs) to reduce the vulnerability of containments to severe accident challenges. For BWRs with Mark I containments, the following improvements were identified:

- Alternative water supply for drywell spray/vessel injection,
- Enhanced reactor pressure vessel depressurization system reliability,
- Implementation of Revision 4 of the BWR Owners Group EPGs, and
- Installation of a hardened vent.

Alternative water supply for drywell spray/vessel injection: Emergency Operating Procedures (EOPs) in the HCGS instruct the operator to inject Station Service Water (SSW) or the fire water into the RPV if all other injection methods fail to provide adequate cooling. The licensee has addressed this issue in baseline CET and sensitivity analyses. The licensee considers the possibility of injecting SSW and fire water into the RPV in node INJ of the CET. Water flowing from the service water pumps through a 36 inch pipe delivers SSW to the Safety Auxiliary Cooling System (SACS) loop. A 6 inch pipe taps off from the 36 inch pipe, and service water is delivered to the RPV through this pipe. Operator actions for injecting service water and fire water are proceduralized through specific EOPs. The fire water systems available for injection into the RPV are the HCGS fire protection system, a cross-tie with the Salem Generating Station (SGS) fire protection system, or a fire truck. Operator actions required include opening a few valves, and installation of a hose to flange adapter. Thus, injection of water from the alternate sources into the RPV, is modelled in the IPE submittal.

In addition, sensitivity study # 1 (page 4.7-16 of the submittal) discusses the alignment of service water system to allow injection through drywell spray system. However, the impact of this alignment on the overall results is rather weak, since the dominant accident sequences are the station blackout sequences. Realizing this, the licensee considered the use of fire protection system (see page 4.7-23 of the submittal) as a source of alternate spray injection. Fire protection system is independent of AC power, and only requires realignment of some valves. Sensitivity study # 15 considers the use of the fire protection system as a source of coolant injection. Results show that the early high and early medium-high releases are completely eliminated, and the conditional probability of intact containment is increased.

Enhanced reactor pressure vessel depressurization system reliability: The licensee performed thermal hydraulic analyses to determine the feasibility of procedural changes for inhibiting and initiating ADS. No ADS procedural changes were recommended based on these analyses. In addition, methods for improving the reliability of ADS were not considered; nor were analyses performed to study the impact of increased reliability of ADS upon the CDF and containment performance.

Implementation of Revision 4 of the BWR Owners Group EPGs: The licensee has implemented the Revision 4 of the BWROG's emergency guidelines as a part of the HCGS EOPs.

Hardened Vent: A hardened vent system has also been installed in the HCGS plant.

E.5 Vulnerabilities and Plant Improvements

The submittal screened for vulnerabilities by seeking sequences and initiating events that contribute inordinately to CDF with respect to (a) other Hope Creek core damage sequences or contributing events, or (2) in comparison to other sequences or events for other nuclear plants as determined from published risk assessments.

The single most significant sequence in the Hope Creek IPE submittal is a total loss of offsite power sequence. A significant contributor to CDF for this sequence is the loss of switchgear or Class 1E panel room HVAC. Loss of HVAC for this room was identified as a "vulnerability". A recovery procedure was developed by the licensee to supply alternate ventilation to the two rooms. The new procedure is stated to be capable of eliminating the "vulnerability". No other vulnerabilities were identified. No hardware modifications, based on the back-end analyses, have been planned, based on the results of the IPE.

A qualitative criterion was used to determine vulnerabilities related to containment performance. The criterion was that the HCGS containment performance results were compared to the results from similar BWRs, and a vulnerability was identified if the HCGS results were significantly different from the other plants. Relatively large frequencies and conditional probabilities of the large-early and medium-early releases were observed by this comparison. The licensee stated that the large frequencies were not attributable to any HCGS containment features, but rather to the dominant contribution of the station blackout sequences to the CDF. However, the licensee concluded that even though the conditional probability of early containment failure given a station blackout was high, it was still comparable to results of other PRAs (e.g., NUREG-1150 analyses for Peach Bottom and Grand Gulf).

Station blackout sequences contribute to 74% of the CDF in the HCGS IPE submittal, and these sequences were dominated by the failure of the diesel generators due to inadequate cooling. The licensee concluded that model conservatism in the SSW and SACS system analyses could be removed. The design basis requires that two out of two SSW pumps are needed for diesel generator cooling; however, new calculations showed that functioning of one pump was sufficient for the successful operation of the SSW loop. Similarly, each SACS loop could function with one pump if the operators are successful in manipulating SACS loads to allow the operation. Existing procedural guidance was deemed sufficient for the operators to perform the required actions.

As a result of crediting the modified success criterion for the SACS and SSW systems, the station blackout CDF was found to be reduced from 3.38×10^{-5} per reactor year to 2.33×10^{-6} per reactor year. The overall CDF was found to be reduced from 4.58×10^{-5} per reactor year to 1.29×10^{-5} per reactor year. The frequency of early and late containment failure, and early-high and early-medium releases are all expected to decline, but would still remain greater than 10^{-6} per reactor year.

E.6 Observations

The following are the major findings of the Hope Creek IPE submittal:

- The Hope Creek CDF is dominated by long-term station blackout sequences, and they contribute to 73.8% of the total CDF. The long-term station blackout sequences lead to a high conditional probability of early containment failure and thereby, radiological releases. This is due to the unavailability of AC power in many sequences to operate the ECCS, the alternate coolant injection systems and drywell sprays.

- Transients without decay heat removal contributing to about 5% of the total CDF are also important, because they often lead to early containment failure and high radiological releases.

- The early containment failure frequency for the Hope Creek plant is driven by the unavailability of coolant injection in many sequences. Early containment structural failure is predicted to occur for 55.7% of the CDF. Long term station blackout sequences contribute to approximately 88% of the frequency of early containment failure. Approximately 84% of this mode of containment failure is due to drywell shell melt-through. Coolant injection into the Reactor Pressure Vessel (RPV) and onto the drywell floor, can prevent containment failure for most of these sequences.

- Drywell shell melt-through is treated in the submittal as containment rupture. Radionuclide retention in the primary system and the reactor building is assumed to be small, and suppression pool bypass is assumed. The Filtration, Recirculation, and Venting System (FRVS) located inside the reactor building is assumed to fail after containment rupture. Accordingly, large radiological releases are estimated.

- Due to the high frequency of drywell shell meltthrough and the low radionuclide retention characteristics of this failure mode, radiological releases are relatively high in a significant fraction of the accident sequences. The frequency of an early high release (defined as releases occurring within 2 hours after vessel breach, with magnitudes larger than 6% of the inventory of iodine and tellurium) is 21% of the total

CDF, and the frequency of an early medium-high release (defined as releases with magnitudes larger than 6% of the inventory of iodine and 0.1 to 6% of tellurium, or releases larger than 6% of the inventory of tellurium and 0.1 to 6% of iodine) is an additional 4%.

- Late containment failure occurs in an additional 18% of the sequences. Long term station blackout sequences contribute to 74% of the frequency of late containment failure. The containment does not fail for approximately 20% of the total CDF, with venting taking place in approximately half of these cases. Venting is almost always from the wetwell, and through the hardened vent system installed at HCGS.

- The frequency of early-high and early-medium releases are 9.42×10^{-6} per reactor year and 6.14×10^{-6} per reactor year, respectively.

- The core damage frequency and the radiological release characteristics are expected to be improved by reducing the frequency of long-term station blackout and by increasing the probability of AC power recovery. As a result of crediting the modified success criteria for the SACS and SSW systems, the station blackout CDF was found to be reduced from 3.38×10^{-5} per reactor year to 2.33×10^{-6} per reactor year. The overall CDF was found to be reduced from 4.58×10^{-5} per reactor year to 1.29×10^{-5} per reactor year. The frequency of early and late containment failure, and early-high and early-medium releases, are all expected to decline.

The important points of the submittal-only technical evaluation of the Hope Creek IPE back-end analysis are summarized as follows:

- Through the Hope Creek IPE submittal, the licensee demonstrates a good understanding of the impact of severe accidents on containment failure and radiological releases.

- The treatment of phenomenological issues in the CET is very detailed, and makes use of results from NUREG-1150 analyses, and more recent, NRC-sponsored research. However, the results of the back-end analyses for containment failure and radiological releases are comparable with NUREG-1150 analyses for Peach Bottom.

- The recommendations of the Containment Performance Improvement (CPI) program have been partially addressed in the Hope Creek IPE submittal. The licensee has not considered the impact of enhanced reliability of ADS. All other recommendations have been addressed.

- The licensee has identified a loss of switchgear room cooling and panel room cooling as a "vulnerability"). A procedure consisting of alternate ventilation for the rooms, have been proposed to eliminate this vulnerability. The licensee also notes that the substantial contribution of long term station blackout sequences to the CDF is due to a conservatism introduced into the treatment of the design of the SSW and SACS systems. After the modification of the success criteria for the SACS and SSW systems, the CDF due to LT-SBO sequences is reduced by a factor of 3.5. This change, together with the use of proposed procedure to eliminate the vulnerability due to loss of switchgear room cooling would have the potential to significantly alter the Level II insights; hence, the licensee should evaluate the impact of these modifications upon the back-end analyses and results.

- The licensee has identified important insights from the back-end analyses, but makes no subsequent use of these insights. One important insight is that the use of alternate injection into the containment through drywell sprays can significantly impact the probability of containment failure and the magnitude and timing of radiological releases. The licensee stated that there are no plans for the use of alternate injection into the containment (e.g., using fire protection system). Another important insight is the impact of the recovery of AC power upon radiological releases. The licensee should evaluate the insights obtained from the sensitivity studies (for example, see page 4.7-26 of the submittal) and develop procedural and/or hardware modifications that could potentially improve the HCGS containment performance.

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NOMENCLATURE

ADS	Automatic Depressurization System
ATWS	Anticipated Transient Without Scram
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners Group
CCI	Core Concrete Interactions
CDF	Core Damage Frequency
CRD	Control Rod Drive
CET	Containment Event Tree
CHR	Containment Heat Rejection
CPI	Containment Performance Improvement
DC	Direct Current
DCH	Direct Containment Heating
DF	Decontamination Factor
ECCS	Emergency Core Cooling Systems
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
ESF	Engineered Safety Features
EVSE	Ex-Vessel Steam Explosion
FPS	Fire Protection System
FRVS	Filtration, Ventilation and Recirculation System
GE	General Electric
GL	Generic Letter
GKA	Gabor, Kenton, and Associates, Inc.
HCGS	Hope Creek Generating Station
HPME	High Pressure Melt Ejection
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation, and Air-Conditioning
IPE	Individual Plant Examination
ISLOCA	Interfacing Systems Loss of Coolant Accident
IVSE	In-Vessel Steam Explosion
LOCA	Loss of Coolant Accident
LOSP	Loss of Off-Site Power
LPCI	Low Pressure Coolant Injection
LT-SBO	Long Term Station Blackout
MAAP	Modular Accident Analysis Program
MSIV	Main Steam Isolation Valve
NRC	Nuclear Regulatory Commission
PDS	Plant Damage State
PSE&G	Public Services Electric & Gas
PRA	Probabilistic Risk Assessment
RAPA	Reliability And Performance Associates
RCS	Reactor Coolant System

NOMENCLATURE (Continued)

RHR	Residual Heat Rejection
RPV	Reactor Pressure Vessel
SAIC	Science Application International Corporation
SACS	Station Auxiliary Cooling System
SBO	Station Black-Out
SORV	Stuck-Open Relief Valve
SRV	Safety Relief Valve
SSW	Station Service Water
TER	Technical Evaluation Report
TW	Loss of Decay Heat Removal

1. INTRODUCTION

This Technical Evaluation Report (TER) documents the results of the "submittal-only" review of the Hope Creek IPE Back-End submittal [1], based on the following review objectives set forth by the NRC:

- 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 weaknesses of the IPE submittal,
- To provide a preliminary list of questions based on this limited review, and
- To complete the IPE Evaluation Data Summary Sheet.

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 Hope Creek 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 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 data 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 through 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 are 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 December 1994. A list of questions and requests for additional information was developed to help resolve issues and concerns noted in the examination process, and was forwarded to the licensee. The final TER is based on the information contained in the IPE submittal [1], and the licensee responses to the NRC Requests for Additional Information (RAIs) [10].

1.2 Containment Analysis

The Hope Creek plant is a General Electric Company BWR/4-251 (251 inch diameter vessel) with Mark I containment. The Hope Creek plant is very similar to the Peach Bottom plant. The plant and containment features are described in Section 4.1 and in Figures 4.1-1 through 4.1-16 of the submittal. The drywell is a steel pressure vessel with a spherical lower portion 68 ft. in diameter, and a cylindrical upper portion 40 ft., 6 inches in diameter. The steel vessel is enclosed in a reinforced concrete biological shield. The drywell internal design pressure is 56 psig and the maximum temperature is 340°F. The top of the drywell is capped with a double gasketed, bolted head. The drywell has four drain sumps, two of which are located in the RPV pedestal floor. The total volume of the two sumps is 210 cubic feet.

Table 1 Comparison of Hope Creek and Peach Bottom Plant and Containment Design Features that Contribute to The Progression of Severe Accidents

Feature	Hope Creek	Peach Bottom
Power Level, MW(t)	3,293	3,293
Volume of Suppression Pool, m^3	3,364	3,480
Free Volume of Drywell, m^3	5,040	4,502
Volume of Wetwell Air Space, m^3	4,037	3,738
Volume of Drywell Sump(s), m^3	6.0	6.1
Mass of UO_2, kg	165,671	159,400
Mass of ZrO_2, kg	74,844	65,491
S.Pool Water Vol./Power, m^3/MW(t)	1.0	1.1
Containment Volume/Power, m^3/MW(t)	2.75	2.5

Table 2 Comparison of Containment Capacities

Parameter	Hope Creek	Peach Bottom
Containment Design Pressure	0.49 MPa (56 psig)	0.49 MPa (56 psig)
Failure Pressure	0.93 MPa (120 psig)	1.09 MPa (150 psig)

Tables 1 and 2 provide a comparison of containment features between the Hope Creek plant and the Peach Bottom plant, another GE plant with a Mark-I containment [2]. It can be seen that the containment features are comparable between the two plants. In addition, the following plant-specific features are important for accident progression in the Hope Creek plant:

- A hardened containment vent pipe 12 inches in diameter originates from the top of the torus and terminates 150 feet above the ground level, outside the secondary containment. The vent can be opened remotely from the control rooms or locally.

Local actuation can be accomplished in the absence of electric power. The operators can cycle the vent between 60 psig and 25 psig to maintain the containment pressure below 60 psig.

- A Filtration, Recirculation, and Venting System (FRVS) is located inside the reactor building that maintains the reactor building at negative pressure and capable of filtering fission products from the secondary containment, once a release occurs.

- A motor-driven fire pump and a standby diesel driven fire pump comprise the Fire-Protection System (FPS). The FPS can be aligned to RHR and service water systems. A procedure exists in the Hope Creek plant for the cross-connect, but no credit is taken in the IPE submittal. The FPS system is located outside the secondary containment, and can also be connected to the Salem FPS.

Table 1 shows that the ratios of containment free volume-to-power and the suppression water volume-to-power are comparable between the Hope Creek and the Peach Bottom plants. Other aspects such as fuel and zirconium masses are also comparable between the two plants.

2. CONTRACTOR REVIEW FINDINGS

The present review compared the Hope Creek IPE submittal to the recommendations of Generic Letter (GL) 88-20, according to 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 Review and Identification of IPE Insights

2.1.1 Completeness and Methodology

The IPE submittal contains a substantial amount of information in accordance with the recommendations of GL 88-20 and NUREG-1335.

The methodology employed in the Hope Creek IPE submittal for the back-end evaluation is clearly described, and the IPE is logical and consistent with GL 88-20. A small event tree/large fault tree approach was employed in the front-end analysis, producing a moderately small number (81) of core-damage accident sequences having a frequency in excess of 10^{-10} per reactor year (internal cut-off frequency in HCGS IPE). Of the 81 core-damage sequences produced in the Level I analysis, 17 dominant sequences are screened in for evaluation in the Level II study, based on the screening criteria of NUREG-1335 (Section 2.1.6). The 17 dominant core-damage sequences, used as initiators for the Level II analysis, together account for more than 95% of the CDF, and they include all core-damage sequences with a frequency greater than 10^{-7} per reactor year. The Level II binning criteria for the HCGS IPE were developed based on the criteria applied in the NUREG-1150 analysis of Peach Bottom Unit 2. The submittal uses a large event tree method for containment analyses.

Probabilistic quantification of severe accident progression involved development of a large event tree. The CET includes a number of nodes that relate to the availability of containment systems. In addition, the CET also includes a probabilistic evaluation of severe accident phenomenology. The results of the CET analyses lead to a number of end-states, which were in turn binned into a small number of release categories, based on similarities in source term characteristics (magnitude and timing of releases). A source term algorithm was used to evaluate the magnitude of the radiological releases.

2.1.2 As-Built/As-Operated Status

The IPE team performed a walk-through of the Hope Creek reactor building. The submittal states that plant walk-downs were performed by the PRA analyst performing the systems analysis. No other information on walkdowns could be found. The review suggests that a focussed post-analysis confirmatory walk-down be performed.

2.1.3 Licensee Participation and Peer Review of IPE

The HCGS IPE effort was a cooperative utility-consultant effort, with most of the work being performed by PSE&G staff. PSE&G utilized contractor support to train PSE&G staff in PRA methods, with emphasis on "hands-on" training. PSE&G provided overall coordination of the HCGS IPE through its PRA group, provided engineers (including senior reactor operators) to support the study, performed portions of the PRA tasks in-house, and reviewed the study results. For the back-end analysis, PSE&G provided one full-time engineer to directly support the effort and one half-time engineer to develop and utilize a HCGS-specific MAAP model.

Contractor assistance on the HCGS IPE is summarized as follows. Halliburton NUS provided training and the primary leadership for the baseline quantification tasks. SAIC provided technical direction in areas of event-tree development, special initiating events analysis, human reliability analysis, and cutset editing and analysis. SAIC also performed an HCGS-specific containment bypass analysis, and provided technical expertise and training for the IPE back-end analysis. Gabor, Kenton and Associates were used by PSE&G to review a portion of the MAAP parameter file. ERIN provided the primary leadership and performed evaluations for the plant-specific Interfacing Systems LOCA analysis. ABB Impell performed a containment capacity analysis for the HCGS IPE.

A consultant from Reliability And Performance Associates (RAPA) provided technical direction for independent review of the HCGS IPE. The remainder of the review team consisted of three PSE&G engineers. The focus of the in-house review team was on methodological correctness, completeness, compliance, consistency, and reasonableness of the IPE. Accuracy of bottom-line numbers was not verified in the review. The review process resulted in over 200 questions being asked by the review team concerning the IPE. It is significant to note that the originally planned submittal date for the IPE was postponed by PSE&G due to issues arising from the independent review.

Suggestions to improve the HCGS IPE, that resulted from the independent review, are provided in Table 5-1 of the submittal. Based on the IPE submittal, it is difficult to ascertain the extent to which PSE&G has implemented, or plans to implement, each of these suggestions.

The review team elucidates a significant concern pertaining to the results of the HCGS back-end analysis. Specifically, the reported frequencies of early large release and early medium release of 9.42×10^{-6} per reactor year and 6.14×10^{-6} per reactor year, respectively, are high (both as absolute levels and levels relative to the reported core-damage frequency). The review team states that the IPE selections regarding designation of release levels were "arbitrary" and "implicitly defined 'large release,' a term that the industry has yet to reach agreement on." However, even though the review team notes that the IPE reported frequency of an early large release is 10 times higher than the 10^{-6} per reactor year (suggested) guideline for regulatory implementation, the review team simply accepts the results because no industry-agreed standard exists.

2.2 Containment Analysis

This section provides a review of PDS binning, CET analyses, release category definitions, severe accident analyses, and the containment structural analyses in the submittal.

2.2.1 Front End/Back End Dependencies

The results of the front-end event trees in the HCGS IPE are accident sequences and their frequencies. A small event tree/large fault tree approach was employed in the front-end analysis, producing a moderately small number (81) of core-damage accident sequences having a frequency in excess of 10^{-10} per reactor year (internal cut-off frequency in HCGS IPE). The frequencies of these 81 sequences ranged from the highest of 3.27×10^{-5} per reactor year down to the lowest of 1.4×10^{-10} per reactor year.

Of the 81 core-damage sequences produced in the Level I analysis, 17 dominant sequences are screened for evaluation in the Level II analysis, based on the screening criteria of NUREG-1335 (Section 2.1.6). Because the Interfacing Systems LOCA (ISLOCA) frequency (1.7×10^{-9} per reactor year) falls outside of the screening criteria, ISLOCA (and containment bypass) sequences were not considered in the Level II analysis. The 17 dominant core damage sequences, used as initiators for the Level II analysis, together account for more than 95% of the CDF, and they include all core-damage sequences with a frequency greater than 10^{-7} per reactor year.

The Level II binning criteria for the HCGS IPE were developed based on the criteria applied in the NUREG-1150 analysis of Peach Bottom Unit 2. These binning criteria are stated to include all system and component states that could affect accident progression and containment response at HCGS. The Level II PDS binning criteria are shown in Table 4.3-2 of the IPE, and include the following characteristics that serve to define the accident bins:

- INITIATING EVENT TYPE
  - Transient
  - LOCA
- REACTOR SUBCRITICALITY STATUS
  - Critical
  - Subcritical
- OFFSITE AC POWER STATUS
  - LOSP
  - No LOSP

- ONSITE AC POWER STATUS

    - SBO
    - No SBO

- DC POWER STATUS

    - Loss of all DC
    - No Loss of DC

- SRV STATUS

    - One, or more, SORV
    - No SORVs

- PRESSURE SUPPRESSION SYSTEM STATUS

    - SRV tailpipe rupture
    - No SRV tailpipe rupture

- REACTOR VESSEL PRESSURE STATUS

    - RPV at high pressure at onset of core damage, and depressurization is not possible
    - RPV at high pressure at onset of core damage, but depressurization is possible
    - RPV at low pressure

- INJECTION STATUS

    - Injection is not recoverable after core damage
    - Some injection, with either high or low pressure systems, is recoverable after CD

- CRD STATUS

    - CRD Is not injecting into vessel, and is not recoverable
    - CRD Is not injecting into vessel, but is recoverable
    - CRD has failed
    - CRD is injecting

- RHR STATUS

    - RHR is not available nor is recoverable
    - RHR is available or Is recoverable

- CONTAINMENT SPRAY STATUS

  - Sprays are not available nor recoverable
  - Sprays are available or recoverable

- CONTAINMENT VENTING STATUS

  - Containment vented
  - Containment venting is possible
  - Containment venting is not possible

- CONTAINMENT LEAKAGE STATUS

  - Small containment isolation failure or leakage prior to core damage
  - Large containment isolation failure or leakage prior to core damage

- LEAKAGE LOCATION

  - Drywell
  - Drywell head
  - Wetwell

- FILTRATION, RECIRCULATION, AND VENTILATION SYSTEM STATUS

  - FRVS is not available nor recoverable
  - FRVS is available or recoverable

- CORE-DAMAGE TIMING

  - Core damage occurs within 1 hour
  - Core damage occurs within 1 to 4 hours
  - Core damage occurs within 4 to 24 hours
  - Core damage occurs after 24 hours

Based on these binning characteristics, vectors defining the status of core-damage sequences were developed. Sequence categories, or sub-sequences (among the 17 dominant sequences), were defined based on the fraction of sequence cutsets falling into a given category (i.e., pertaining to a given combination of binning-criteria states). When applied to cutsets defining the 17 dominant core-damage sequences, a total of 35 unique vectors (representing 35 unique combinations of binning criteria) were developed. These 35 unique combinations of binning criteria define 35 unique plant damage states. Table 4.3-3 of the HCGS IPE provides a characterization of Level I sequences used for Level II sequence binning.

The IPE submittal says that all 35 PDSs are represented explicitly (i.e., fully) in the Level II analysis. Hence, plant states defined by the binning criteria should be accurately accounted for in the back-end analysis. However, for reporting purposes, the IPE provides results of the Level II analysis (including the resulting containment matrix) only in terms of 5 initiator classes (LT-SBO, Other Transients, LOCAs, TW, and ATWS). This characterization of Level II initiators is also used when presenting containment event trees (CETs). Although this representation of the results is certainly a useful approach for summarization purposes, it is surprising that this is the only characterization used to subsequently portray results in terms of (quasi) plant damage states. It is considered important that the IPE submittal present plant damage states, and plant damage state frequencies, in terms of the 35 unique PDSs based on the 35 unique vector combinations of binning criteria.

Table 4.3-4 of the submittal provides a list of initiator categories and their CDFs for use in the back-end analyses. The dominant contributors to core damage are long-term SBO sequences (73.8%), followed by transient sequences (14.8%), LOCAs (6.7%), special initiators (3.1%), and ATWS sequences (1.6%). ISLOCA and containment bypass sequences are not included in the Level II analysis. In addition, there were no short-term SBO or small break LOCAs that met the 1×10^{-7} screening criterion that was applied to the Level I sequences. A more complete description of fractions of plant damage states associated with each initiator category cannot be obtained until Table 4.3-3 of the IPE is modified to explicitly define binning vectors and the fractional portions of each of the 17 dominant sequences belonging to the unique binning combinations.

Station blackout sequences contribute to 74% of the CDF in the HCGS IPE submittal, and these sequences were dominated by the failure of the diesel generators due to inadequate cooling. The licensee concluded that model conservatism in the SSW and SACS system analyses could be removed. The design basis requires that two out of two SSW pumps are needed for diesel generator cooling; however, new calculations showed that functioning of one pump was sufficient for the successful operation of the SSW loop. Similarly, each SACS loop could function with one pump if the operators are successful in manipulating SACS loads to allow the operation. Existing procedural guidance was deemed sufficient for the operators to perform the required actions.

As a result of crediting the modified success criterion for the SACS and SSW systems, the station blackout CDF was found to be reduced from 3.38×10^{-5} per reactor year to 2.33×10^{-6} per reactor year. The overall CDF was found to be reduced from 4.58×10^{-5} per reactor year to 1.29×10^{-5} per reactor year.

2.1.2.2 Containment Event Tree Development

Probabilistic quantification of severe accident progression for the probabilistically significant accident bins was performed using CETs. The methodology employed in the Hope Creek IPE submittal involved "linked" event trees, where linking implies that there are common events among the event trees. The CET has thirteen top event nodes, which are further developed using subtrees. Some of the subtrees have additional "sub-subtrees" for specific phenomena. The CET is evaluated using the EVNTRE code developed by Sandia National Laboratory (SNL) for NUREG-1150. In a qualitative sense, the CET methodology is similar to the large event tree method used for the NUREG-1150 back-end analyses, and it appears that the use of subtrees is primarily for the purpose of illustration. The branch point probabilities in the subtrees (referred to as basic events) are assigned by the IPE analysts. CET quantification is performed using results from plant-specific MAAP analyses and published results using other codes such as CONTAIN and MELCOR. Results from NUREG-1150 analyses were used to supplement the available data, and to envelop the ranges of phenomenological uncertainty. Results from more up-to-date literature were used for the treatment of uncertainties in drywell shell melt-through.

Figures 4.5-1A through E of the submittal depict the HCGS CETs for each of the five level II initiator (long-term station blackout, medium/large LOCAs, transients, transients with loss of decay neat removal, and ATWS sequences). The same CET is used for the five accident initiators. The top events (considered in the supporting subtrees) are described below.

Event 1 Vessel Depressurized (DP)

The possibility of depressurization of the RPV prior to vessel breach is considered by this node. Success implies that RPV pressure is reduced either through the capability of the operator to depressurize the reactor or through a phenomenological condition that induces RPV depressurization. Conversely, transient accident sequences in which the RPV is at low pressure (through opening the SRVs) may be repressurized if the ADS valves reclose upon high containment pressure.

For accident sequences with the RPV at high pressure (and Low Pressure Coolant Injection (LPCI) pumps deadheaded), depressurization of the RPV can lead to recovery of coolant makeup. In addition, high RPV pressure could exacerbate containment challenges at vessel breach (such as high pressure melt ejection), and these challenges to containment integrity are removed. This event node directly impacts the likelihood of the subsequent CET event nodes related to in-vessel recovery and early containment loads. Depressurization is possible by operator action to open the SRVs after core damage, or by a stuck-open relief valve during the cycling of SRVs. Recovery of AC power is not relevant to this event, except for the case where the containment pressure is too high to prevent SRV operation. If containment venting is successful, then the SRVs can be opened to depressurize the RPV.

The submittal assigns a basic event probability of 0.052 for the failure of the operator to depressurize after core damage, subsequent to the recovery of AC power. In addition, a very low probability of 2×10^{-5} is assigned for the SRVs to reopen after closure due to high containment pressure.

Event 2 Injection Recovered In-Vessel (INJ)

This question considers the recovery of coolant injection after core degradation and prior to vessel breach. The potential for arresting core melt and subsequent thermal failure of the RPV is also considered by this node. It considers high-pressure injection systems, the possibility of functional low pressure injection systems once the RPV is depressurized, and the recovery of injection systems given AC power recovery in station blackout scenarios.

For sequences where the containment is initially intact and the RPV is at pressure, core damage might be induced by lack of coolant makeup due to failure of the high pressure injection systems. However, the low pressure injection systems may be available, but coolant injection is prevented by conditions that preclude pump operation (i.e., RPV pressure exceeding the shutoff head). Once the RPV is depressurized, coolant injection would most likely be recovered. Success at this branch facilitates core recovery in-vessel, implies fission product scrubbing, and may prevent ultimate containment failure. For sequences involving low RPV pressure (i.e., lrge LOCAs with failure to provide adequate coolant makeup), success at this node is not likely, as implied by the accident sequence definition.

The sub-tree used for the determination of the split fractions for this top event is provided in Figure 4.5-3 of the submittal. First, the availability or the recovery of AC power is considered. Here, it should be noted that the submittal did not do a plant-specific analysis for the recovery of AC power, but used the recovery curves presented in NUREG-1150 for the IPE analyses. For long-term station blackout sequences, the conditional probability of AC power not being recovered early is assigned a value of 0.605. Given the availability or recovery of ECCS, first, the ability of CRD systems to arrest core damage is addressed. The submittal does not provide the conditional probability for core damage arrest by CRD flow alone. For a depressurized sequence, the availability of low pressure injection systems are considered. The availability of alternate injection systems (condensate and service water) is also considered. The conditional probabilities of core damage arrest are provided in the next node.

Event 3 Vessel Failure (Vat)

This event addresses recovery of a degraded core within the vessel, which prevents vessel failure. Arrest of core melt within the vessel is considered only to the extent that coolant makeup has been successful in the previous node (INJ). The submittal states that "This requires that core cooling be recovered prior to core blocking (MAAP model) or relocation of molten debris to the lower plenum and thermal attack of vessel head". Therefore, the licensee assumes the primary consideration for successful in-vessel recovery is the time available from incipient core degradation to the point of non-recovery.

The success of in-vessel recovery (see Figure 4.5-4 of the submittal) is dependent on two factors: recovery of injection, and extent of core degradation. If more than 26% of the inventory of the core is relocated at the time of recovery of injection, a basic event probability of 0.3 is assigned for vessel breach. If less than 26% of the core is relocated at the time of recovery of injection, it is assumed that the probability of vessel breach is 0.001. The bases for the assignment of these probabilities, and the choice of the 26% cut-off value for core damage arrest, is not provided in the submittal. More importantly, the extent of core damage for various accident classes (at the time of recovery of injection) is also not provided in the submittal. This is not important in the Hope Creek IPE submittal because the conditional probability of in-vessel recovery is low (due to the dominant contribution of long term station blackout sequences to core damage), but may become important if the station blackout sequences become less dominant. In addition, degraded core coolability due to heat transfer through the vessel to water on the pedestal floor is also not considered as a probable method of prevention of vessel breach.

Event 4 Early Containment Failure (CFE)

This event node addresses containment failure at or around vessel breach. In addition, intentional venting of the containment is treated as a failure mode. This in turn is used to evaluate various fission product decontamination mechanisms and a fractional release of fission products from the primary to the secondary containment. Early containment (structural) failure occurs due to phenomena accompanying vessel failure. Containment leakage (failure size sufficient to prevent further long-term pressure increase and lead to reduction to atmospheric pressure over a period longer than ten to twelve hours) and containment rupture (failure size sufficient to reduce containment pressure to atmospheric within half an hour) are the two possible modes of failure, and are distinctly treated by the fission product retention (EPR) event. This distinction significantly impacts the environmental source term.

Early containment failure is analyzed using the sub-tree depicted in Figure 4.5-5 of the submittal. The containment pressure at vessel breach is evaluated as the sum of the pressure rise due to blowdown (including the effect of energetic events such as DCH) and the containment pressure prior to vessel breach. In addition, containment failure due to Fuel-Coolant Interactions (FCI) and drywell shell melt-through are treated independently using sub-trees. Both the pressure rise and the baseline pressure are divided into low, moderate and high ranges. It is stated in the submittal that all the MAAP calculations for the various accident sequences were tabulated and used to arrive at the above-mentioned pressure ranges, but, the submittal does not provide these ranges.

Containment Pressure Rise

The sub-tree that is used to evaluate the containment pressure prior to vessel breach uses the following dependencies:- ECCS injection to the core, CRD injection, and the amount of in-vessel hydrogen generated. The effect of functional RHR system upon containment pressure is ignored in this sub-tree.

Blowdown

The sub-tree used to evaluate the pressure rise in a containment resulting from the blowdown of a pressurized vessel after vessel breach. The first node in this sub-tree determines whether the accident sequence is at high or low pressure. The next node determines whether vessel breach occurs for this sequence. Both of these nodes were discussed earlier. The next node determines whether a "slump" (defined as collapse of core en-masse) occurs, and a conditional probability of 0.1 is assigned for this basic event. It is not clarified why the core has to collapse en-masse (for subsequent melt ejection) or why a value of 0.1 is assigned for this event. Given that the vessel is at high pressure and core slumps en-masse into the lower plenum, a conditional probability of 0.8 is assigned for High Pressure Melt Ejection (HPME). The pressure rise at vessel breach is also dependent on the availability of water on the pedestal floor. Three ranges of containment pressure rise, namely, low, medium or high are defined.

The methodology used for the treatment of containment pressurization at vessel breach (due to HPME) in the submittal is quite weak. In addition, there is no definition in the submittal for the ranges of containment pressure (high, medium and low), and their conditional probabilities.

Meltthrough

This subtree evaluates the conditional probability of drywell shell melt-through due to contact with debris after it spreads across the drywell floor. Shell melt-through is assumed to be dependent on (1) occurrence of "slump" (massive relocation of core), (2) occurrence of HPME, (3) extent of debris dispersal, and (4) availability of water on the drywell floor. The conditional probabilities of drywell melt-through under dry and wet conditions were obtained from the results of Theofanous, et al. [3]. Even though the intermediate basic event probabilities are not provided, the final results are given and are listed in Table 3 below. The discussion of the phenomena is rather scant, but, the results appear to reflect the licensee's understanding of the phenomena. The results from NUREG/CR-5423 are used, together with some additional results from NUREG/CR-4551, and the quantification is in line with many other recent IPE submittals.

Table 3 Results from the Licensee Evaluation of Containment Failure Due to Shell Melt-through

Dependencies	Conditional Probability of Shell Melt-Through
No Core Slump, No Debris Dispersal, Water on Pedestal Floor	0.0001
No Core Slump, No Debris Dispersal, No Water on Pedestal Floor	0.63
No Slump, Debris Dispersal, No Water on Pedestal Floor	0.51
Core Slump, No HPME or Dispersion, Water on Pedestal Floor	$6x\ 10^{-5}$
Core Slump, No HPME or Dispersion, No Water on Pedestal Floor	0.999
Core Slump, No HPME, Limited Dispersion, No Water on Pedestal Floor	0.6
Core Slump, HPME, No Water on Pedestal Floor	0.79

Steam Explosions

This sub-tree is used to determine the probability of occurrence of a fuel-coolant interaction, and the outcome of this event relative to containment failure. First, the probability of in-vessel steam explosion was assumed to be 0.05, and the overall probability of vessel rupture (and containment failure) is assumed to be 0.0001. These values were obtained from estimates calculated by ERIN for BWR owners group as a part of an EPRI-supported study. In comparison, NUREG-1150 used values of 0.001 for high pressure sequences, and 0.01 for low pressure sequences for the conditional probability of containment failure due to in-vessel steam explosion. Therefore, the conditional probability for IVSE-induced containment failure in the submittal is unsupported.

The sub-tree is also used to treat ex-vessel steam explosion-induced containment failure. The governing parameters are assumed to be the amount of molten material, and the availability of water on the pedestal floor. With a large mass of molten material available, the submittal assumes that the probability of ex-vessel FCI is 0.5. With a small mass of molten material available, the submittal assumes that the probability of ex-vessel FCI is 0.1. The conditional probability of containment failure for the large mass case is 0.1, and 0.01 for the case when only a small mass of molten material available for participation in FCI. These values are said to be based on engineering judgement, and are superficial.

Event 5 Early Release to Pool (EPOOL)

This event considers suppression pool bypass and the possibility that drywell-to-wetwell vacuum breakers are stuck open. Only those releases that occur after core damage and at vessel failure are considered in this event. A conditional probability of 0.001 is assigned for the vacuum breakers being stuck open.

Event 6 Drywell Sprays Operate (DWSpry)

This event addresses the functioning of the drywell sprays after vessel failure. Drywell sprays can provide aerosol removal, perform containment pressure suppression by heat removal (if the suppression pool fails to function), and act as a source of water that can potentially cool ex-vessel debris. This event considers the effect of harsh severe accident environment (such as draining of the suppression pool after wetwell failure and supporting equipment failure in harsh environment). The effect of late AC power recovery on drywall spray function is also considered. The conditional probabilities of spray failure due to harsh environments and spray recovery due to AC power recovery, could not be determined from the submittal.

Event 7 Injection Provided to Debris Following Vessel Failure (L-INJ)

This event addresses the issue of coolant injection to the debris following vessel breach. Late coolant injection can potentially cool the debris, and scrub the fission product releases. This event considers the availability of injection systems, AC power, and the probability of actuation given AC power recovery after vessel failure. Injection actuation late in the course of an accident scenario is possible for two cases: (1) injection systems available and AC power is recovered late in station blackout scenarios, and (2) low pressure injection systems available, but RPV pressure is high until vessel failure. Coolant injection can potentially quench the debris. However, there are significant disagreements within the scientific community on the potential for an overlying pool of water to cool the debris. Experiments performed to date do not support coolability. Successful late injection can possibly reduce the revolatilization fission product release fraction and can possibly cool an ex-vessel debris bed, precluding late releases from core-concrete interactions. Availability of the diesel-driven fire water system as a mode of coolant injection to debris is not considered in the base case analyses.

Event 8 Coolable Debris Forms Ex-Vessel (DCOOL)

This event considers the possibility of termination of the core melt progression subsequent to vessel breach. Success branch at this node means that a coolable debris bed is formed, terminating concrete attack, precluding fission product releases during core-concrete interactions, and implying that containment overpressure challenges from noncondensible gas generation are ceased. Containment integrity may be maintained over the long term if heat removal and sufficient coolant makeup is available.

The sub-tree is depicted in Figure 4.5-9 of the submittal. The significant phenomenological uncertainty is considered in the DCOOL node of the sub-tree, where the conditional probabilities for debris bed coolability for various conditions are assigned. Table 4 shows the conditional probabilities for debris coolability as provided in the submittal. It is heartening to note that the licensee has assigned a significant conditional probability of non-coolable debris with water addition, even though the plant-specific MAAP analyses show otherwise.

Table 4 Licensee Evaluation of the Conditional Probabilities of Core Debris Coolability

Dependencies	Conditional Probability of Non-Coolable Debris
No Core Slump, No Debris Dispersal, Water on Drywell Floor	0.1
Core Slump, No Debris Dispersal, Water on Pedestal Floor	0.5
No Core Slump, Debris Dispersal, Water on Pedestal Floor	0.05
No Core Slump, No Dispersion, Water Added After Vessel Breach	0.5
Core Slump, No Dispersion, Water Added After Vessel Breach	0.9
Core Slump, Dispersion, Water Added After Vessel Breach	0.1
Debris Dispersal, No Water on Pedestal Floor	0.99

Event 9: Late Containment Failure (CFL)

Loss of containment integrity in the long term, after vessel breach and core-concrete interactions is addressed by this event. Event CFL includes such events as overpressure failure of the primary containment, containment failure due to high temperatures, and basemat melt-through. Success depends on the recovery of systems that establish heat transfer from the core debris.

Most plant-specific calculations showed that late containment failure occurred at the CRD hatch or the drywell head. The sub-tree for this event is shown in Figure 4.5-10. Thermal failure of seals is also possible. Another additional, failure mode identified by the licensee is sump ablation (because the bottom of the outer drywell sumps (which connect to the in-pedestal sumps) is only 7.4 inches from the drywell shell). The conditional probabilities of late containment failure mode in the submittal or the following:

Probability of containment failure given that water is added and debris is cooled = 0.01
Probability of containment failure given that water is not added and debris is not cooled = 0.95
Probability of containment failure given that water is added and debris is not cooled = 1.0
Probability of sump failure given that debris is not coolable, and no other mode of failure = 0.9

Event 10 Late Release to Pool (LPOOL)

This event is similar to event EPOOL. Suppression pool scrubbing and mitigation of the magnitude of fission products released from the debris in the late phase of the accident is addressed in this event.

Event 11 Fission Product Retention (RR)

This event (and the following event) is related to the binning of source term releases. The effects of drywell spray, water on the drywell floor, and containment integrity on fission product retention within the containment are considered. The containment failure mode (leak or rupture) and the location of that failure is also considered. Leaks enhance fission product retention by increasing the extent of deposition within containment. This is due to the greater residence time for fission products when the rate of flow is reduced by the small size of failure. It is assumed that the added retention by the leak is assumed to be comparable to water scrubbing of core-concrete interaction releases or deposition of airborne materials due to drywell spray operation, but no basis for this assumption is provided.

Event 12 Secondary Containment Retention (RB)

This event is included in the CET to characterize the impact of mitigation due to the secondary containment following containment failure. This event considers the type of containment failure, whether or not a hydrogen burn occurs in the reactor building, and the availability of active fission product removal by the FRVS in the reactor building.

Event 13 Vent

This is a summary event that gives the capability to differentiate between containment failure and intentional venting. Thus, containment venting is not considered as a form of containment failure. Sequences where venting is successful at preventing either early or late containment failure can have a significantly lower environmental fission product releases than those where containment structural failure occurs.

The overall methodology employed in the Hope Creek IPE submittal for CET analysis is well organized. The Hope Creek CET includes most of the relevant severe accident phenomena applicable to BWRs with Mark I containments. Most of the details of the quantifications are provided. However, the technical bases of some of the uncertain basic events is weak.

2.2.3 Containment Failure Modes and Timing

A plant-specific evaluation of the structural capacity of the HCGS Mark I containment was conducted for the HCGS IPE, and is summarized in Section 4.4 of the IPE submittal. This analysis focuses on integrity of the primary containment. Capability of the secondary containment (i.e., reactor building, including both domed and rectangular concrete structures) to resist internal pressure loadings is very low. MAAP analyses performed by PSE&G have shown that the secondary containment is likely to fail a few hours after primary containment failure, usually as a result of pressure increase caused by flammable gas combustion in one or more compartments of the reactor building. Pressure loads are easily transmitted throughout the reactor building pathways (doors, stairways, steam vent, steam tunnel, etc.); in addition, a number of ready pathways exist for radionuclide releases from the reactor building to the outside environment (particularly through louvered blow-out panels that open under a low differential pressure of 1.5 psid). Although the FRVS (if functional) may postpone failure of the secondary containment, margin against such failure (given primary containment failure) is low; therefore, the major effect of the reactor building in the back-end analysis is to provide additional retention of fission product releases prior to being released to the environment.

The HCGS IPE submittal states that the full range of potential loads, including static pressurization, elevated temperatures, and dynamic loads resulting from energetic phenomena (e.g., fuel coolant interaction) have been considered. In actuality, although static pressurization and elevated temperature effects are considered, the submittal gives only a superficial treatment of dynamic loads. The dynamic loads and explosion resistances are not based on a HCGS-specific analysis. Due to inerting, the submittal states that the possibility of combustion in the containment is remote, and is thus ignored in the back-end analysis. It should be noted that the possibility exists for a significant steam explosion which may challenge the primary containment with dynamic loading [4]. Plant-specific dynamic analyses are required to adequately assess the response of the primary containment subject to dynamic explosion loads. Hence, the HCGS IPE submittal is not considered to provide a very meaningful treatment of dynamic loads from potential energetic phenomena.

Structural integrity of the HCGS primary containment (drywell and wetwell) under over-pressure-related severe-accident load conditions was performed by PSE&G's contractor, ABB Impell. Fragility assessments were based on quasi-static load analyses. The locations examined in this failure analysis included the following:

- Drywell shell;
- Drywell head flange;
- Vent lines from the drywell to the suppression pool;
- Torus shell (wetwell);
- Drywell equipment hatch;
- Drywell personnel airlock;
- Control rod drive (CRD) removal hatch; and
- Piping Penetrations

In Figure 4.4-1, the submittal demonstrates that mean failure pressures for each of the preceding 9 failure locations were computed for 5 different temperature conditions, ranging from 200°F to 1000°F. A comparison between Figures 4.4-1 and 4.4-2, however, suggests that either the results in Figure 4.4-1 are actually median (not mean) failure pressures, or that the mean failure pressures were mistakenly used to characterize the median failure pressures in Figure 4.4-2.

Plant-specific MAAP calculations showed that, prior to vessel breach, temperature in the primary containment is relatively low, but after vessel breach temperature rises rapidly to a higher level. Temperatures of 200°F and 600°F are used to roughly characterize these two conditions. Fragility curves for the 9 locations, for temperature conditions of 200°F and 600°F, respectively, are provided in Figures 4.4-2 and 4.4-3. From these figures, values of β for primary containment fragility at 200°F and 600°F are estimated to be roughly in the vicinity of 0.26 and 0.40, respectively. These values are considered to be somewhat low; they may not adequately reflect realistic uncertainty in determination of a failure condition. In fact, for CET quantification, the IPE submittal (p. 4.4-2) does use a higher failure probability than that developed in the containment capacity analysis, in order to account for "potential uncertainties arising from the analysis approach" of the containment capacity study.

Median failure pressures for the primary containment were estimated to be 120 psig and 21 psig, respectively, for temperatures of 200°F and 600°F. At 200°F, failure of the drywell head flange is assessed in the IPE to dominate containment primary failure, whereas at 600°F, the predominant failure mode is assessed to be failure of the CRD removal hatch. The failure mode at 200°F (leading to rapid "O"-Ring seal degradation) is judged in the IPE to be associated with rapid depressurization, similar to a containment rupture, whereas at 600°F, the failure mode is classified as a leak.

The MAAP code was the principal tool in the IPE used for predicting pressure and temperature histories within the containment. These predictions were used, together with the containment capacity/fragility assessments, to predict the timing and location of containment failure. Assessment of the probability of containment failure for CET quantification also considered drywell shell melt-through. Basemat melt-through is a significant consideration at HCGS because the bottom of the outer drywell sumps (which connect to the within-pedestal sumps) is only 7.4 inches from the drywell shell.

2.2.4 Containment Isolation Failure

In the Hope Creek IPE submittal, containment isolation failure was analyzed, but was found to be an insignificant contributor to containment failure. No other details are provided. The IPE submittal concludes that the probability of either containment bypass or containment isolation failure for HCGS was sufficiently small that sequences involving either one had frequencies below the NUREG-1335 screening criteria. Consequently, these sequences were not addressed in the back-end analysis. A plant-specific ISLOCA analysis was conducted for the IPE, producing ISLOCA frequency below the screening criteria.

2.2.5 System/Human Response

Twentytwo basic events involving operator failure are identified in the Level 2 analysis (Table 4.6-1 of the submittal) that impact CETs. These basic events, and their probabilities, are summarized below:

- ALT-FL-1: Operator fails to provide flow from alternate systems after vessel failure, given that there was no station blackout, low-pressure ECCS is not available, and alternate injection systems are available. Vessel was not depressurized prior to vessel breach, and thus this is the first opportunity for the operators to use alternate injection. (basic event probability = 0.052).

- ALT-FL-2: Operator fails to provide flow from alternate systems after vessel failure, given that there was no station blackout, low-pressure ECCS is not available, and alternate injection systems are available. Vessel was previously depressurized, but operators failed to use alternate injection source at that opportunity, as required by procedures. (basic event probability = 1.0).

- ALT-FL-3: Operator fails to provide flow from alternate systems, given a station blackout, but with AC power restored late; ECCS is not available. (basic event probability = 0.052).

- ALT-FLOW-1: HEP (Human Error Probability) for failure to align alternate injection systems (CST or SW) during core damage, given that at least one system is available, and because of a prior error of omission was not previously restored. (basic event probability = 1.0).

- ALT-FLOW-2: HEP for failure to align alternate injection systems (CST or SW) during core damage, given that at least one system is available, AC power is restored early, and ECCS is not available. (basic event probability = 0.052).

- CD -> DP - 1: Operator fails to depressurize the RPV after core damage, given that the SRVs are operable in relief mode and DC power is available (i.e., prior error of omission). (basic event probability = 1.0)

- CD -> DP - 3: Operator fails to depressurize the RPV after core damage, given that the SRVs are operable in relief mode, but DC power was previously unavailable and AC power is then restored. (basic event probability = 0.052)

- CRD-FLOW: Operator fails to provide CRD flow to the vessel, given that AC power is restored, and CRD pumps are operable. (basic event probability = 0.052).

- CRD-L-1: Operator fails to provide CRD flow to the debris, given that there is not a station blackout and CRD pumps are operable. (Operators have previously failed to provide CRD flow, as required). basic event probability = 1.0).

- CRD-L-2: Operator fails to provide CRD flow to the debris, given that there is a station blackout, but AC power is restored late, and CRD pumps are operable. (basic event probability = 0.052).

- SPRY-E: Operator fails to initiate drywell sprays early (as required) to depressurize the containment. (basic event probability = 0.052).

- SPRY-L-1: Operator fails to initiate drywell sprays to cool core debris, given that the vessel has breached and drywell sprays are available. (basic event probability = 1.0).

- SPRY-L-2: Operator fails to initiate drywell sprays to cool core debris after vessel breach, given a SBO with AC power restored early and sprays available. (basic event probability = 1.0).

- SPRY-L-3: Operator fails to initiate drywell sprays to cool core debris, given vessel failure, SBO, late restoration of AC power, and spray system available. (basic event probability = 1.0).

- VENT-E/AC: Operator fails to vent containment prior to RPV failure, given that containment pressure is sufficient to require venting, and AC power and vent valves are available. (basic event probability = 0.052).

- VENT-E/DW: Operator vents containment early through the drywell and not through the wetwell, given that the containment has been vented. (basic event probability = 0.05).

- VENT-E/MAN: Operator fails to manually vent the containment early, given that containment pressure is elevated, AC power is not available, and vent valves are available. (basic event probability = 0.31).

- VENT-L/AC: Operator fails to vent containment late, given that containment pressure is sufficient to require venting, and AC power and vent valves are available. (basic event probability = 0.052).

- VENT-L/DW-INJ: Operator vents containment late through the drywell and not through the wetwell, given that the containment has been vented and injection systems are operating. (basic event probability = 0.95).

- VENT-L/DW-DRY: Operator vents containment late through the drywell and not through the wetwell, given that the containment has been vented and injection systems are not operating. (basic event probability = 0.05).

- VENT-L/MAN-L: Operator fails to manually vent the containment late, given that containment pressure is elevated, AC power is not available, and vent valves are available. This event applies to LT-SBO sequences. (basic event probability = 0.31).

- VENT-L/MAN-S: Operator fails to manually vent the containment late, given that containment pressure is elevated, AC power is not available, and vent valves are available. This event applies to ST-SBO sequences. (basic event probability = 0.31).

2.2.6 Radionuclide Release Categories and Characterization

The results of the CET analyses lead to an extensive number of end-states, which are in turn binned for source term analyses. Outcomes of the CETs are classified into a small number of release categories, which are based on similarities in accident progression and source term characteristics.

As discussed in Section 4.7.2.1 of the submittal, only two characteristics were identified as having the greatest impact on fission product release at Hope Creek, and they are the following:

- Timing of Release

Late	-	Beyond 2 hours after vessel breach
Early	-	From the time of accident initiation until 2 hours after vessel breach

- Magnitude of Release

High (1)	-	Greater than 6% of iodine inventory and 6% of tellurium inventory
Medium-High (2)	-	Greater than 6% iodine release and 0.1 - 6% of tellurium release, or greater than 6% tellurium release and 0.1 - 6% of iodine release

Medium (3)	-	0.1 - 6% of iodine or tellurium release
Low (4)	-	0.001 to 0.1% of iodine and tellurium inventory
Low-low (5)	-	Less than 0.001% of iodine and tellurium inventory

The choice of the use of the magnitude of iodine and tellurium releases for release category definitions was made in the submittal based upon a review of NUREG-1150 results.

To estimate the source terms, an algorithm was developed and integrated with the EVNTRE code through the use of "User Function" capability available in that code. The source term code uses sequence-dependent Release Fractions (RFs) and Decontamination Factors (DFs) within correlations that calculate the net fractional radionuclide releases for five species for several accident sequences. Only five radionuclide groups were evaluated for the IPE, namely, noble gases, iodine, cesium, tellurium, and strontium. Strontium is assumed to be representative of all refractory species such as strontium, barium, beryllium, magnesium, and ruthenium. The algorithm used to quantify the source term releases for the HCGS IPE is based on the approach developed in NUREG-1150, and is described below.

A release Fraction is defined as the mass fraction of the available material in a given fission product group that is released from the core (debris) and becomes available for release. Release fractions were defined for each fission product, and for three processes, namely, in-vessel release from the fuel (R_E) ex-vessel release from MCCI (R_{MCCI}), and revolatilization releases (R_{REV}) from radionuclides that are initially released from the fuel, but are trapped on the reactor coolant system surfaces. The total releases to the environment is the sum of the in-vessel releases, releases during core-concrete interactions, and releases due to revolatilization, and is given as:

$$R_{tot}\ (i) = \left[R_E\ (i) * \frac{F_{vent}}{DF_{ESC}(i)} \right] + \left[\left[R_{MCCI}\ (i) + R_{REV}\ (i) \right] * \frac{F_{vent}}{DF_{LSC}(i)} \right] \qquad (1)$$

Where,

$R_E(i)$	=	In-vessel release fraction.
$R_{MCCI}(i)$	=	Core-concrete interaction release fraction.
$R_{REV}(i)$	=	Revolatilization release fraction.
F_{vent}	=	Fractional release of fission products from the primary containment to the secondary containment. Values less than one are possible if containment vent recloses after opening.

DF_{ESC} = Secondary containment decontamination factor for early in-vessel releases.

DF_{LSC} = Secondary containment decontamination factor for MCCI and revolatilization releases.

The individual release terms are defined as follows:

In-vessel Releases:

$$R_E(i) = \frac{RF_{IV}(i)}{DF_{IV}} \tag{2}$$

CCI Releases:

$$R_{MCCI}(i) = [1 - RF_{IV}(i)] * \frac{RF_{MCCI}(i)}{DF_{MCCI}(i)} \tag{3}$$

Revolatilization Releases:

$$R_{REV(i)} = \left[RF_{IV}(i) * \left(1 - \frac{1}{DF_{VSL}(i)} \right) \right] * \frac{RF_{REV}(i)}{DF_{REV}(i)} \tag{4}$$

The release fractions are defined as follows:

$RF_{IV}(i)$ = In-vessel release fraction for each radionuclide group, (i).

$RF_{MCCI}(i)$ = Core-concrete interaction release fraction for each radionuclide group, (i).

$RF_{REV}(i)$ = Revolatilization release fraction for each radionuclide group, (i).

Deposition mechanisms act on this released material to limit its ultimate release to the environment. Decontamination Factors (DFs) defined for the radionuclide group in the above equations account for the reduction in airborne mass of fission products by various deposition mechanisms. Mathematically, the DF is the ratio of initial mass of fission product species available for release from a given volume to the mass that remains available after the decontamination mechanism has taken effect. Decontamination factors were defined for each fission product group for RCS and vessel deposition, natural deposition in the containment and spray decontamination, as follows:

$DF_{EPOOL}(i)$ - Early suppression pool scrubbing decontamination factor.

$DF_{LPOOL}(i)$ - Late suppression pool scrubbing decontamination factor.

$DF_{VSL}(i)$ - Primary system decontamination factor.

$DF_{ECONT}(i)$ - Early containment deposition decontamination factor (for in-vessel releases).

$DF_{ESPR}(i)$ - Early spray decontamination factor (for in-vessel releases).

$DF_{ERB}(i)$ - Early decontamination factor for the secondary building.

$DF_{LCONT}(i)$ - Late containment deposition decontamination factor (for ex-vessel and revolatilization releases).

$DF_{LSPR}(i)$ - Late spray decontamination factor (for ex-vessel and revolatilization releases).

$DF_{LRB}(i)$ - Late decontamination factor for the secondary building.

Decontamination factors (used in Equations 2 through 4) were defined for in-vessel releases, core-concrete-interaction releases, and revolatilization releases, as follows:

Decontamination factor for in-vessel releases:

$$DF_{IV}(i) = DF_{ECONT}(i) * DF_{VSL}(i) * DF_{ESPY}(i) * DF_{EPOOL}(i) \tag{5}$$

Decontamination factor for CCI releases:

$$DF_{MCCI}(i) = DF_{LCONT}(i) * DF_{LSPY}(i) * DF_{LPOOL}(i) \tag{6}$$

Decontamination factor for revolatilization releases:

$$DF_{REV}(i) = DF_{LCONT}(i) * DF_{LSPY}(i) * DF_{LPOOL}(i) \tag{7}$$

Evaluation of the radiological releases requires a knowledge of RFs and DFs. Some values were obtained from HCGS MAAP 3.0B calculations. They were supplemented by results obtained from the NUREG-1150 expert elicitation process as documented in NUREG/CR-4551.

In-Vessel Releases

MAAP analyses were performed for several accident sequences, and in-vessel release fractions were obtained for several species. Table 4.7.2 (page 4.7-29 of the submittal) provides the values of the release fractions used in the submittal. The in-vessel releases calculated by the submittal were larger for the noble gases, iodine and cesium. The NUREG-1150 values were used for tellurium, and a slightly lower value is used for strontium. A comparison of the NUREG-1150 values, and the release fractions used in the IPE submittal, is shown in Table 5.

These estimates are also consistent with recent MELCOR calculations [5]. Decontamination factors for the vessel fission product transport were obtained from the MAAP calculations. A value of 2 for cesium and a value of 3 for other species appear to have been used. These are comparable with the NUREG-1150 values.

Decontamination Factors for Suppression Pool and Drywell Sprays

A decontamination factor of 1000 is used for suppression pool scrubbing. This value is based on MAAP calculations for scrubbing of fission product aerosols through the T-quenchers MAAP prediction for T-quencher DFs vary from 120 to 1000, and it is not apparent why the larger value was used.

Decontamination factors for early and late sprays were not available from MAAP calculations. The submittal uses a DF value of 10 for both, early and late sprays, and this estimate appears to be reasonable.

Revolatilization Releases

It is assumed that only water in the coolant loops (due to late injection) affects revolatilization releases. The revolatilization release fraction for the two cases, with and without late injection used in the submittal are provided in Table 6. The values for revolatilization releases without coolant injection for the submittal were obtained from MAAP calculations. The release fractions (for no coolant injection cases) are larger than the NUREG-1150 values, and this is attributed by the licensee to higher primary system temperatures predicted by the MAAP code. These estimates are consistent with values calculated by the MELCOR code [5]. For sequences with late coolant injection, release fractions were obtained from NUREG-1150 analyses.

Table 5 Comparison of In-Vessel Releases From the Fuel: Peach Bottom NUREG-1150 Analyses and Hope Creek IPE

	Peach Bottom	Hope Creek
Group	(NUREG-1150)	IPE [1]
NG	0.9	1.0
I	0.69 -0.74	0.99
Cs	0.59	0.99
Te	0.14 - 0.15	0.15
Sr	4E-3 - 6.4E-3	0.001

CCI Releases and Decontamination by Water Pool

Table 7 shows a comparison of MCCI release fractions, between the values used in the IPE submittal and the NUREG-1150 study. The values used in the IPE were based on MAAP calculations, and they are comparable with the NUREG-1150 estimates. The release fraction for strontium is smaller for the submittal, and the licensee states that " lower values for SR release are justified based on recent experiments in SNL and ANL". However, references for these studies are not cited.

Containment Decontamination Factors

Containment decontamination factors were not available for early releases from the MAAP calculations. NUREG/CR-4551 provided estimates varying from 1.3 to 3.5 for early releases, and these values were used. Containment decontamination factors for late releases were obtained from MAAP results. Values between 1.4 to 3.0 were used. These estimates are not consistent, and larger values of DFs should be used for late releases. In comparison, NUREG-1150 analysts provided larger values for DFs, between 15 and 24.

Reactor Building Decontamination Factors

Reactor building decontamination factors based on MAAP calculations were not available for early releases. The decontamination factors calculated for late releases to the reactor building

Table 6 Comparison of Revolatilization Releases: Peach Bottom NUREG-1150 Analyses and Hope Creek IPE

	Peach Bottom	Hope Creek
Without Late Coolant Injection	NUREG-1150	IPE [1]
I	0.115	0.7
Cs	0.051	0.7
Te	0.0	0.0
Sr	0.0	0.0
With Injection		
I	0.03	0.03
Cs	0.001	0.001
Te	0.0	0.0
Sr	0.0	0.0

Table 7 Comparison of MCCI Releases: Peach Bottom NUREG-1150 Analyses and Hope Creek IPE

	Peach Bottom	Hope Creek
Group	NUREG-1150	IPE [1]
I	1.0	0.95
Cs	1.0	0.95
Te	0.23 - 0.36	0.4
Sr	0.027 - 0.052	0.01

were also used for early releases. Two values were used, namely, a value of 1.5 for "low deposition" and a value of 10 for "high deposition". High deposition factors are assigned for accident sequences that involve leakage from the containment and no hydrogen burns in the reactor building. For all other sequences, the lower value was used. In contrast, NUREG/CR-4551 provides estimates varying from 1.35 to 4.

In summary, the release fractions and decontamination factors used in the submittal appear to be reasonable, and in some cases, conservative. However, the actual magnitude of the source terms have not been provided in the submittal. Instead, source terms have been classified into high, medium or low based on the magnitude of cesium, iodine and tellurium releases. There are no quantitative estimates provided for the various source term bins.

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," or "any functional sequences that contribute to a containment bypass frequency of 10^{-7} per reactor year," should be reported. The submittal provides a list of the reportable sequences in Section 7.1.1.1 of the submittal based on the requirements outlined in Section 7.1.1 of the submittal. However, the requirements defined by the submittal do not take into account the NRC reporting criteria for releases. Tables 4.7-3 through 4.7-12 provide the release frequencies for the five accident initiators. It is difficult to obtain the reportable sequences from these tables. The reason is that the high releases have magnitudes greater than 6% (of iodine inventory), but it is not known whether the releases are greater than 10% of the iodine inventory. The first and third sequences in the long term station blackout PDS (Table 4.7-4 of the submittal) have release frequencies larger than 10^{-6} per reactor year and releases larger than 6% of the iodine and tellurium inventory. However, it is to be noted that all the core damage sequences are reported in Table 7.1.1 of the submittal. The dominant contributor to early releases (79%) and late releases (68%) is the TeEDG sequence. This sequence is initiated by loss of offsite power followed by failure of emergency diesel generators, failure of HPCI and RCIC due to battery failure after four hours, and core damage occuring within two hours after battery failure.

Two points are to be noted. Unlike several other IPE submittals, the licensee has correctly classified the timing of releases, including the releases at or around reactor pressure vessel breach into the "early" category. Sequences corresponding to approximately 62% of the CDF, entail early containment failure. In addition, the classification of releases based on the 6% threshold is quite conservative, particularly in relation to other IPE submittals. Sequences corresponding to 28.4% of the CDF lead to large releases (greater than 6% of iodine or tellurium inventory).

2.3 Quantitative Assessment of Accident Progression and Containment Behavior

2.3.1 Severe Accident Progression

MAAP-BWR 3.0B, Revision 8.1 was the principal tool used to analyze postulated severe accidents at Hope Creek. The MAAP input file is not provided as a part of the submittal, but the input file and the supporting calculations are documented in a PSE&G (internal) document. The licensee has correctly recognized that the MAAP-BWR code is not capable of modelling severe accident phenomena such as DCH, liner melt-through and steam explosions. With regards to ex-vessel core debris coolability, the submittal also notes that "with the default modelling parameters values, MAAP assumes that the heat transfer from the core debris to the coolant is the flat plate critical heat flux". In addition, it is also noted that "at this heat transfer rate (~ 1 MW/m^2), the core debris is always coolable when water is present", and "this result is not validated by comparison to experiments". Because of this, other views of ex-vessel debris core coolability were considered in HCGS containment analyses.

A number of simulations were performed using the MAAP code for five accident sequences (LOCA, long-term station blackout, transient, ATWS and TW). However, the submittal does not clearly list the simulations performed, results of the simulations, and details of the sensitivity analyses. It appears that the recommended sensitivity analyses in the EPRI document [6] were not performed.

2.3.2 Dominant Contributors to Containment Failure

Table 8 shows a comparison of the conditional probabilities of the containment failure modes provided in the Hope Creek IPE submittal, together with the results of the IPE submittals for the Fitzpatrick and Vermont Yankee plants, as well as the NUREG-1150 study for Peach Bottom [7]. Table 9 provides the C-matrix for the Hope Creek plant. From a review of Table 6, it is seen that the only major difference between Hope Creek and the Peach Bottom plant, is that the Hope Creek IPE has a higher conditional probability of late containment failure.

In the Hope Creek IPE submittal, the dominant contributor to CDF is the long-term station blackout sequences (73.8%), followed by transients (9.8%), LOCA sequences (6.7%), and TW sequences (5%). In contrast, for the Peach Bottom NUREG-1150 analyses, the dominant contributor to CDF is the LOSP sequences(46%), followed by ATWS sequences (42%), transients and LOCA sequences.

Table 8 Containment Failure as a Percentage of Internal Events CDF: Comparison of Hope Creek IPE Results to Other BWR Mark I IPEs and Peach Bottom NUREG-1150 Results

Containment Failure	Fitzpatrick IPE	Peach Bottom NUREG-1150	Vermont Yankee IPE	Hope Creek IPE
CDF (per year)	1.9×10^{-6}	4.5×10^{-6}	4.3×10^{-6}	4.58×10^{-5}
Early Failure	60	56	48	62^{+}
Bypass	NA	NA	1	< 1
Late Failure	26	16	24	28^{++}
Intact	3	18	27^{*}	10^{*}
No Vessel Breach	11	10	NA	NA

NA - Not Available
* - Includes Both Intact Containment and No Vessel Breach Cases
\+ - Includes the Following Breakup: Structural Failure = 55.7%; Venting = 6.4%
++ - Includes the Following Breakup: Structural Failure = 17.6%; Venting = 10.4%

Table 9 Hope Creek C-Matrix

	Release Category						
	Initiator (% of Total CDF)	Late Containment Failure	Late Containment Venting	Early Containment Failure	Early Containment Venting	Intact	Total
Plant	LT-SBO (74)	0.164	0.104	0.633	0.053	0.046	1.0
	TW (5)	0.208	0.03	0.693	0.046	0.024	1.0
Damage	Transients (13)	0.064	0.187	0.248	0.12	0.381	1.0
	LOCA (7)	0.011	0.011	0.078	0.094	0.806	1.0
State	ATWS (2)	0.181	0.181	0.256	0.112	0.27	1.0

Table 10 Conditional Probabilities of Early Containment Failure Modes in the HCGS IPE

Initiator (PDS)	Drywell Melt-through	FCI-Induced Failure	Early Overpressure
Long Term SBO	0.535	3.29E-4	0.095
TW	0.639	0	0.058
Transients	0.233	2.41E-3	0.22
LOCA	0.071	5.28E-3	0.001
ATWS	0.142	1.29E-3	0.11
Total	0.469	0.0007	0.09

Table 11 Conditional Probabilities of Late Containment Failure Modes in the HCGS IPE

Initiator (PDS)	Late Overpressure	Overtemperature	Sump Melt-through
Long Term SBO	4×10^{-4}	0.089	0.074
TW	7×10^{-4}	0.142	0.064
Transients	3×10^{-4}	1.2×10^{-3}	0.116
LOCA	5×10^{-4}	0.433	0.113
ATWS	4×10^{-4}	5.3×10^{-3}	0.071
Total	0.0004	0.102	0.075

The additional difference between the Peach Bottom and the Hope Creek studies is the failure pressure used for late containment failure. At a temperature of 600°F, a failure pressure of 21 psig is indicated for Hope Creek, whereas, for the Peach Bottom NUREG-1150 analysis, a failure pressure of 120 psig was used. A large number of accident sequences will end up in late containment failure, if vessel breach is not prevented.

There are small containment failure mode differences between the two plants. The breakup of containment failure probabilities into failure modes in the Hope Creek submittal is presented in Tables 10 and 11. The largest contributor to early containment failure (by failure mode) is drywell liner melt-through, followed by overpressure. The largest contributor to late containment failure is late overtemperature, followed by sump ablation. The identification of sump ablation as a possible failure mode, together with the higher conditional probabilities of late overtemperature failure (compare these results with those for Peach Bottom shown in Figure S-3, page S.17 of Reference [7]), together lead to the calculation of larger conditional probabilities of late containment failure in the IPE submittal.

2.3.3 Characterization of Containment Performance

The MAAP code was used to calculate several accident progression parameters (e.g., peak pressures from quasi-static overpressurization at RPV failure). Several important phenomena, such as ex-vessel steam explosions, Mark I shell failure, and direct containment heating, were treated probabilistically using results from NUREG-1150, NUREG/CR-5423, and other sources. The results from the baseline CET analyses are illustrated in Tables 8 and 9. In addition to the base case CET analyses, a number of sensitivity calculations have been performed as a part of the IPE submittal. Sensitivity analyses have been performed for the following parameters:

1. Use of Service Water for Drywell Sprays: In all accident sequences, water supply for drywell sprays was assumed to be available. There was a small increase in the conditional probability of no containment failure (from 12 to 18%), and the conditional probability of early high and early medium-high releases was reduced from 34% to 26%.

2. Operator Action to Use Sprays in the Late Phase of the Accident: In the base CET analyses, it was assumed that the temperature and pressure inside the containment was beyond the spray operating limit. The assumption is conservative, and the sensitivity analyses assume that late spray initiation was always successful. However, the effect the overall frequency of releases was negligible.

3. Sprays Always Available and Always Used Late: The two previous cases were combined in this sensitivity case. The most significant effect was the reduction of early medium-high release frequency from 14% to 0%.

4. No Drywell Shell Melt-through: In this study, the drywell shell melt-through probability was set to zero. The total conditional probability of early releases was reduced from 62% to 29%, and the probability of early/high releases was reduced from 21% to 8%. However, the conditional probability of late failure was increased from 20% to 53%.

5. No Early AC Power Recovery: No credit is given for early AC power recovery in this sensitivity case. The conditional probability of early containment failure was found to increases from 69% to 75% for long-term station blackout sequences.

6. AC Power Always Available: AC power recovery is assured in the early phase of the long-term station blackout, and all ESFs can be used. The conditional probability of early high releases is reduced from 21% to 4%, and the early medium-high releases is reduced from 14% to 13%.

7. Ex-Vessel Core Debris Not Coolable: The conditional probability is reduced to zero. The early medium-high release frequency was increased, while the early medium and early low releases were reduced.

8. Ex-Vessel Core Debris Always Coolable: The conditional probability of core debris coolability is increased to one. The release categories were shifted to one lower level (early medium-high releases were reduced to early medium releases, and so on).

9. Injection Systems Never Fail Due to Harsh Environment: No impact on releases was noted.

10. Alternate Injection Systems Never Fail Due to Harsh Environment: No impact on releases was noted.

11. Drywell Vent: Venting was assumed always to be from the drywell. There is a slight shift from the low release to medium release category.

12 - 14 FRVS Efficiency: The impact of FRVS (scrubbing efficiency and availability) upon releases was found to be weak.

15. Use of Fire System to Ensure Late Spray Availability: A non AC-power independent source was identified; in addition, the drywell spray valves were assumed to be open. Thus spray availability was assured. The net effect of sprays was to completely eliminate the high and medium-high releases. A higher frequency of intact containment was calculated.

The sixteenth sensitivity study, opening of torus vents prior to vessel failure, was not conducted. Only limited discussions based on engineering judgement are presented.

In summary, containment analyses performed for the IPE submittal are detailed, and all important modes of containment failure have been considered.

2.3.4 Impact on Equipment Behavior

Section 4.6.2.31 (page 4.6-8) of the submittal provides a discussion of the effects of harsh environmental conditions upon the survivability of ESFs. The Hope Creek CET considers the survivability of ECCS, alternate injection systems, and drywell sprays due to harsh environmental conditions in the reactor building. The baseline analyses used the values obtained from NUREG-1150 results for the Peach Bottom plant for equipment survivability probabilities. However, there are differences between the Peach Bottom and Hope Creek reactor building designs:

- FRVS is used to filter, recirculate and ventilate aerosols within the secondary containment, whereas the Peach Bottom plant uses a once-through ventilation without recirculation. The recirculation mode of the FRVS is stated to prevent significant pressurization or heatup in any part of the reactor building.

- The HCGS secondary containment is dome shaped and has a slightly larger volume than Peach Bottom reactor building.

- The blowout panels in the torus room and steam tunnel will open at 1.5 psid and relieve pressure to the atmosphere.

- The HPCI and RCIC rooms are equipped with blowout panels that will open to the torus room at 0.25 psid.

- The Hope Creek design is more compartmentalized, and the licensee feels that harsh environments in one part of the Hope Creek reactor building will not be communicated to other areas.

- Some elements of the alternate cooling system in HCGS are located in the turbine building rather than the reactor building. Thus, it appears that alternate injection systems such as condensate pumps will continue to function even after containment failure.

The licensee believes that the use of the Peach Bottom equipment survivability probabilities for the HCGS plant is conservative, given the differences between the two designs.

A sensitivity analysis was performed to investigate the assumption that alternate injection systems will function even after containment failure. However, the impact of assumed survivability of the ESFs upon source term releases (frequency and magnitude) was weak.

2.4 Reducing the Probability of Core Damage or Fission Product Release

2.4.1 Definition of Vulnerability

The submittal screened for vulnerabilities by seeking sequences and initiating events that contribute inordinately to CDF with respect to (a) other Hope Creek core damage sequences or contributing events, or (2) in comparison to other sequences or events for other nuclear plants as determined from published risk assessments.

The single most significant sequence in the Hope Creek IPE submittal is a total loss of offsite power sequence (see page 7-2 of the submittal). A significant contributor to CDF for this sequence is the loss of switchgear or Class 1E panel room HVAC. Loss of HVAC for this room was identified as a "vulnerability". A recovery procedure was developed by the licensee to supply alternate ventilation to the two rooms. The new procedure is stated to be capable of eliminating the "vulnerability". Other vulnerabilities were not identified.

A qualitative criterion was used to determine vulnerabilities related to containment performance. The criterion was that the HCGS containment performance results were compared to the results from similar BWRs, and a vulnerability was identified if the HCGS results were significantly different from the other plants. Relatively large frequencies and conditional probabilities of the large-early and medium-early releases were observed by this comparison. The licensee stated that the large frequencies were not attributable to any HCGS containment features, but rather to the dominant contribution of the station blackout sequences to the CDF. However, the licensee concluded that even though the conditional probability of early containment failure given a station blackout was high, it was still comparable to results of other PRAs (e.g., NUREG-1150 analyses for Peach Bottom and Grand Gulf). It was already noted that, as a result of crediting the modified success criterion for the SACS and SSW systems, the station blackout CDF was found to be reduced from 3.38×10^{-5} per reactor year to 2.33×10^{-6} per reactor year. The overall CDF was found to be reduced from 4.58×10^{-5} per reactor year to 1.29×10^{-5} per reactor year. The frequency of early and late containment failure, and early-high and early-medium releases, are all expected to decline.

2.4.2 Plant Modifications

No hardware modifications, based on the back-end analyses, have been planned.based on the results of the IPE.

2.5 Responses to the Recommendations of the CPI Program

Generic Letter 88-20, Supplement Numbers 1 and 3 [8,9] identified specific Containment Performance Improvements (CPIs) to reduce the vulnerability of containments to severe accident challenges. For BWRs with Mark I containments, the following improvements were identified:

- Alternative water supply for drywell spray/vessel injection,
- Enhanced reactor pressure vessel depressurization system reliability,
- Implementation of Revision 4 of the BWR Owners Group EPGs, and
- Installation of a hardened vent.

The recommendations of the CPI program have not been addressed directly in the Hope Creek IPE submittal. However, some of the individual recommendations have been addressed in the submittal, and they are discussed below:

Alternative water supply for drywell spray/vessel injection: The licensee has addressed this issue in baseline CET and sensitivity analyses. The licensee considers the possibility of injecting service water and fire water into the RPV in the node INJ of the CET. Water flowing from the service water pumps through a 36 inch pipe delivers SSW to the SACS loop. A 6 inch pipe taps off from the 36 inch pipe, and service water is delivered to the RPV through this pipe. Operator actions for injecting service water and fire water are proceduralized through specific EOPs. The fire water systems available for injection into the RPV are the HCGS fire protection system, a cross-tie with the Salem Generating Station (SGS) fire protection system, or a fire truck.

operator actions required include opening a few valves, and installation of a hose to flange adapter. Thus, injection of water from the alternate sources into the RPV, is modelled in the IPE submittal.

In addition, sensitivity study # 1 (page 4.7-16) discusses the alignment of service water system to allow injection through drywell spray system. However, the impact of this alignment on the overall results is rather weak, since the dominant accident sequences are the station blackout sequences. Realizing this, the licensee considered the use of fire protection system (see page 4.7-23 of the submittal) as a source of alternate spray injection. Fire protection system is independent of AC power, and only requires realignment of some valves. Sensitivity study # 15 considers the use of the fire protection system as a source of coolant injection. Results show that the high and early medium-high releases are completely eliminated, and the conditional probability of intact containment is increased. This is an important finding, however, the licensee has no plans to implement the use of fire protection system as an alternate source of drywell spray injection.

Enhanced reactor pressure vessel depressurization system reliability: The licensee performed thermal hydraulic analyses to determine the feasibility of procedural changes for inhibiting and initiating ADS. No ADS procedural changes were recommended based on these analyses. In addition, no methods for improving the reliability of ADS was considered; no analyses were performed to study the impact of increased reliability of ADS upon the CDF and containment performance.

Implementation of Revision 4 of the BWR Owners Group EPGs: The licensee has implemented the Revision 4 of the BWROG emergency procedural guidelines as a part of the HCGS EOPs.

Hardened Vent: The hardened vent was discussed in Section 2.1.2. The licensee only states that "the availability of hardened vent system improves the capability of Hope Creek plant to remove decay heat". The licensee believes that if the vent system is opened prior to vessel failure, the magnitude and timing of the releases will be reduced substantially. This conclusion is based on engineering judgement, and not based on severe accident analyses.

2.6 Insights, Improvements and Commitments

The licensee identifies these findings in Section 6.3 based on back-end analyses:

- The long-term station blackout sequences are the dominant contributors to CDF. Due to the dominance of these sequences, the sensitivity studies showed little variations (of the conditional probability of containment failure and release frequencies) to most of the sensitivity parameters considered. This is particularly true of the availability of ESF systems.

- The results are very sensitive to assumptions of AC power recovery. If AC power is always recovered early, then the conditional probability of early high releases is reduced from 21% to 4%, and the late high releases are reduced from 7% to 1.4%. On the other hand, if AC power is never recovered early, the conditional probability of early, high releases are increased to 32%.

- The results are insensitive to the availability of drywell sprays (without altering AC power availability).

- The FRVS is a system unique to Hope Creek plant that can scrub radionuclides in the reactor building. However, the overall results are not significantly impacted by the availability of the FRVS system.

- The results are significantly impacted by two uncertainties in severe accident phenomena, namely, drywell shell melt-through and debris coolability on the pedestal floor. The impact of these two phenomena are discussed in detail in Section 4.7.3 and summarized in Page 6-3 of the submittal.

- The CDF and the radionuclide release characteristics are expected to be improved by reducing the frequency of long-term station blackout and increasing the probability of AC power.

At the time of submission of the IPE documentation, the licensee had committed to revisions in the modelling of two important systems (SACS and SSW systems). The licensee determined that model conservatisms in the SSW and SACS system analyses could be removed. The design basis requires that two out of two SSW pumps are needed for diesel generator cooling; however, new calculations showed that functioning of one pump was sufficient for the successful operation of the SSW loop. Similarly, each SACS loop could function with one pump if the operators are successful in manipulating SACS loads to allow the operation. Existing procedural guidance was deemed sufficient for the operators to perform the required actions.

As a result of crediting the modified success criterion for the SACS and SSW systems, the station blackout CDF was found to be reduced from 3.38×10^{-5} per reactor year to 2.33×10^{-6} per reactor year. The overall CDF was found to be reduced from 4.58×10^{-5} per reactor year to 1.29×10^{-5} per reactor year. The frequency of early and late containment failure, and early-high and early-medium releases, are all expected to decline.

3. OVERALL EVALUATION AND CONCLUSIONS

The back-end portion of the Hope Creek IPE submittal provides a substantial amount of information in regard to the subject areas identified in Generic Letter 88-20 and NUREG-1335. Of the submittals that have been reviewed by ERI, this submittal is one of the most detailed in scope, in containment and source term analyses. The PRA methodology used for the back-end analysis is sound, capable of identifying plant-specific vulnerabilities to release of radionuclide material, and includes all key phenomenological issues. The submittal considers all phenomena of interest to severe accident phenomenology applicable to BWRs with Mark I containments. The treatment of phenomenologic issues in the CET is very detailed, and the IPE makes use of results from NUREG-1150 analyses, and more recent, NRC-sponsored research. The severe accident and source term calculations are detailed. Several minor weaknesses (with regards to their overall impact on the IPE results) exist, and they include the following:

- The licensee does not report the magnitude of source terms, but only classifies them based on their magnitudes.

- The conditional probabilities of in-vessel steam explosions used in the CET are significantly lower than those reported in the literature.

- The treatment of ex-vessel steam explosions and containment failure due to these dynamic loads, is superficial.

- There are some errors in the containment structural analyses, and discrepancies between the failure probabilities calculated by the structural analyses and those used in the CET analyses.

The following are the major findings of the Hope Creek IPE submittal:

- The Hope Creek CDF is dominated by long-term station blackout sequences, and they contribute to 73.8% of the total CDF. The long-term station blackout sequences lead to a high conditional probability of early containment failure and thereby, radiological releases. This is due to the unavailability of AC power in many sequences to operate the ECCS, the alternate coolant injection systems and drywell sprays.

- Transients without decay heat removal contributing to about 5% of the total CDF are also important, because they often lead to early containment failure and high radiological releases.

- The frequency of early containment failure in the HCGS submittal is driven by the unavailability of coolant injection. Early containment structural failure is predicted to occur for 55.7% of the CDF. Long term station blackout sequences contribute to approximately 88% of the frequency of early containment failure. Approximately 84% of this mode of containment failure is due to drywell shell melt-through. Coolant injection into the Reactor Pressure Vessel (RPV) and onto the drywell floor, can prevent containment failure for most of these sequences.

- Drywell shell melt-through is treated in the submittal as containment rupture. Radionuclide retention in the primary system and the reactor building is assumed to be small, and suppression pool bypass is assumed. The Filtration, Recirculation, and Venting System (FRVS) located inside the reactor building is assumed to fail after containment rupture. Accordingly, large radiological releases are estimated.

- Due to the high frequency of drywell shell meltthrough and the low radionuclide retention characteristics of this failure mode, radiological releases are relatively high in a significant fraction of the accident sequences. The frequency of an early high release (defined as releases occurring within 2 hours after vessel breach, with magnitudes larger than 6% of the inventory of iodine and tellurium) is 21% of the total CDF, and the frequency of an early medium-high release (defined as releases with magnitudes larger than 6% of the inventory of iodine and 0.1 to 6% of tellurium, or releases larger than 6% of the inventory of tellurium and 0.1 to 6% of iodine) is an additional 4%.

- Late containment failure occurs in an additional 18% of the sequences. Long term station blackout sequences contribute to 74% of the frequency of late containment failure. The containment does not fail for approximately 20% of the total CDF, with venting taking place in approximately half of these cases. Venting is almost always from the wetwell, and through the hardened vent system installed at HCGS.

- The frequency of early-high and early-medium releases are 9.42×10^{-6} per reactor year and 6.14×10^{-6} per reactor year, respectively.

- The core damage frequency and the radiological release characteristics are expected to be improved by reducing the frequency of long-term station blackout and by increasing the probability of AC power recovery. As a result of crediting modified success criteria for the SACS and SSW systems, the station blackout CDF was found to be reduced from 3.38×10^{-5} per reactor year to 2.33×10^{-6} per reactor year. The overall CDF was found to be reduced from 4.58×10^{-5} per reactor year to 1.29×10^{-5} per reactor year. The frequency of early and late containment failure, and early-high and early-medium releases, are all expected to decline.

The important points of the submittal-only technical evaluation of the Hope Creek IPE back-end analysis are summarized as follows:

- Through the Hope Creek IPE submittal, the licensee demonstrates a good understanding of the impact of severe accidents on containment failure and radiological releases.

- The treatment of most phenomenologic issues in the CET is very detailed, and the study makes use of results from NUREG-1150 analyses, and more recent NRC-sponsored research. However, the results of the back-end analyses for containment failure and radiological releases are comparable with NUREG-1150 analyses for Peach Bottom.

- The recommendations of the Containment Performance Improvement (CPI) program have been partially addressed in the Hope Creek IPE submittal. The licensee has not considered the impact of enhanced reliability of ADS. All other recommendations have been addressed.

- The licensee has identified a loss of switchgear room cooling and panel room cooling as a "vulnerability"). A procedure consisting of alternate ventilation for the rooms, have been proposed to eliminate this vulnerability.

- The IPE submittal reports a substantial contribution of long term station blackout sequences to the CDF. The licensee stated that this result is due to a conservatism introduced into the treatment of the design of the SSW and SACS systems. After the modification of the success criteria for the SACS and SSW systems, the CDF due to LT-SBO sequences is reduced by a factor of 3.5. This change, together with the use of proposed procedure to eliminate the vulnerability due to loss of switchgear room cooling would have the potential to significantly alter the Level II insights; hence, the licensee should evaluate the impact of these modifications upon the back-end analyses and results.

- The licensee has identified important insights from the back-end analyses, but makes no subsequent use of these insights. One important insight is that the use of alternate injection into the containment through drywell sprays can significantly impact the probability of containment failure and the magnitude and timing of radiological releases. However, the licensee has no plans (procedural or plant modifications) for the use of alternate injection into the containment (e.g., using fire protection system). Another important insight is the impact of the recovery of AC power upon radiological releases. The licensee should evaluate the insights obtained from the sensitivity studies (for example, see page 4.7-26 of the submittal) and develop procedural and/or hardware modifications that could potentially improve the HCGS containment performance.

- The recommendations of the Containment Performance Improvement (CPI) program have been partially addressed in the HCGS submittal. The licensee has considered alternate sources of injection to the RPV and implemented revision 4 of the BWROG emergency procedure guidelines. A hardened vent has been installed in the plant. However, the licensee did not consider methods to improve the reliability of the ADS.

4. REFERENCES

1. "Hope Creek Nuclear Power Station Individual Plant Examination," Public Service Electric and Gas Company, April 1994.

2. USNRC, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, Vols. 1 and 2, December 1990.

3. T. G. Theofanous, et al., "The Probability of Liner Failure in a Mark-I Containment," NUREG/CR-5423, August 1991.

4. H. Esmaili, M. Khatib-Rahbar, and U. Shmocker, "A Prababalistic Assessment of Ex-Vessel Fuel-Coolant Interaction Energetics For Light Water Reactors," Tenth Proceedings of Nuclear Thermal Hydraulics, 1994 ANS Winter Meeting, 195-202, 1994.

5. R. Vijaykumar et al., "Simulation of Severe Reactor Accidents: A Comparison of MELCOR and MAAP Computer Codes," Paper Presented at the International Topical Meeting on Probabilistic Safety Assessment, Clearwater Beach, Florida, 1993.

6. Gabor, Kenton & Associates, Inc., "Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP 3.0B," EPRI TR-100167, 1991.

7. "Evaluation of Severe Accident Risks: Peach Bottom, Unit 2," NUREG/CR-4551, Vol. 4, Rev. 1, December 1990.

8. NRC Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR §50.54(f)," Generic Letter 88-20, Supplement No. 1, dated August 29, 1989.

9. NRC Letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities - Generic Letter No. 88-20 Supplement No. 3 - 10 CFR §50.54(f)," Generic Letter 88-20, Supplement No. 3, dated July 6, 1990.

10. Response to Request for Additional Information, Attachment to Letter from E. C. Simpson to the U.S. Nuclear Regulatory Commission, dated November 6, 1995.

APPENDIX A

IPE EVALUATION AND DATA SUMMARY SHEET

BWR Back-End Facts

Plant Name

Hope Creek Nuclear Station

Containment Type

Mark I

Unique Containment Features

None found for the Hope Creek containment. However, plant-specific features of importance to severe accident progression include the following:

- Hardened Torus Vent: The hardened vent is an important mitigating feature for TW sequences.
- A Filtration, Recirculation, and Venting System (FRVS) is located inside the reactor building that maintains the building under negative pressure and is capable of filtering fission products from the building, once a release into the building occurs.

Unique Vessel Features

None found

Number of Plant Damage States

35

Containment Failure Pressure

Median failure pressures are 120 psig and 21 psig, for temperatures of 200°F and 600°F, respectively

Additional Radionuclide Transport and Retention Structures

Reactor building structures

Conditional Probability That the Containment Is Not Isolated

Negligible

Important Insights, Including Unique Safety Features

See Section 4 of the review.

Implemented Plant Improvements

<u>Improvements made prior to the issuance of the Generic Letter 88-20</u>

- Alternative water supply for drywell spray/vessel injection,
- Installation of a hardened vent.
- Alternate sources of injection to the RPV (fire water and service water) were identified and connecting lines to the RPV installed.

C-Matrix

See Table 8 of the review.

HOPE CREEK GENERATING STATION

TECHNICAL EVALUATION REPORT

HUMAN RELIABILITY ANALYSIS

Enclosure 4