APPENDIX B

RIVER BEND UNIT 1 STATION INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(BACK-END)

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FINAL DRAFT

TECHNICAL EVALUATION REPORT OF THE RIVER BEND STATION INDIVIDUAL PLANT EXAMINATION BACK-END SUBMITTAL

FINAL DRAFT

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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 Gulf States Utilities' (GSU) Individual Plant Examination (IPE) for the River Bend Station (RBS) nuclear power plant. The containment analyses were performed as a utility-contractor joint effort by a team consisting of GSU and Halliburton-NUS personnel. EQE Engineering performed the containment fragility analysis. An independent review of the back-end analysis was conducted as follows: Science Application International Corporation (SAIC) reviewed all Level 2 activities; Stone and Webster Engineering Corporation reviewed the containment fragility analysis; Enercon Services provided comments on Level 2 IPE assumptions concerning hydrogen generation.

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

RBS is a single-unit, GE BWR/6 with Mark III steel containment and reinforced-concrete shield building. The plant site is located near Baton Rouge, Louisiana. The design of RBS is similar to Grand Gulf, Perry and Clinton.

The important containment design characteristics at RBS (which differ from the above-mentioned BWR/6 plants with Mark III containments) include the following:

- There is an annular region between the free-standing containment vessel and the outer shield building, which provides for some additional de-contamination of radionuclides before release to the environment. The air volume in this annular region is filtered by the Standby Gas Treatment System (SGTS).

- Unlike other Mark III containments, where containment sprays and drywell sprays are provided for containment cooling, RBS employs fan coolers.

- RBS does not have vacuum breakers or check valves between the containment and drywell atmospheres.

- RBS does not provide for suppression pool make-up via upper fuel-pool dump.

- The access door to the reactor pedestal at RBS is water-tight, and the door is kept closed during plant operation.

- Containment and drywell hydrogen ignitors are powered from separate electrical circuits, providing for independent and preferential activation in these two regions.

- RBS has a 3-inch containment vent line, which is insufficient to prevent containment failure due to over-pressure.

- RBS has a 30-inch containment dome vent which was permanently sealed (welded) after construction.

The significance of these RBS containment features, as modelled in the RBS IPE, are discussed in this review document.

E.2 Licensee's IPE Process

The IPE process for the back-end evaluation of RBS uses inputs and procedures similar to those employed in the NUREG-1150 analysis of Grand Gulf. The Level-2 evaluation starts with the results of the Level-1 analysis; i.e., a description of possible accident sequences, together with estimates of their annual frequencies of occurrence. A Plant-Damage state Logic Diagram (PDLD) is constructed and used to bin core-damage sequences having similar cut sets into given Plant Damage State (PDS) bins. A containment event tree (CET) is constructed and quantified to model accident progression. The CET is supported by decomposition event trees (DETs) for complete modeling and quantification of accident progression. The overall approach is similar to the use of Containment Event Trees (CETs) in NUREG-1150, and the results of this part of the analysis produces Accident Progression Bins (APBs) and their annual frequencies of occurrence. A source term logic diagram (STLD) is then used to describe the progression from APBs to Source Term Categories (STCs). The source term analysis results in definition of source-term categories and their annual frequencies of occurrence, which represent the end products of the back-end analysis.

The licensee performed a number of sensitivity studies for the back-end evaluation. In addition, a meaningful uncertainty analysis was apparently performed, although the details of such are not fully documented in the IPE.

GSU had a major participation in the oversight and execution of the back-end analysis, although NUS clearly had a substantial involvement in technical aspects of the work. The IPE was performed on-site at River Bend, and consultants visited the site whenever plant-specific information concerning the layout of components, maintenance practices, or specific operational information was required. GSU staff involved in the IPE included those having day-to-day familiarity with the plant. The IPE provides only a brief description of the plant walkdown effort, but the description seems to indicate an adequate effort was undertaken (and documented) so as to define the as-operated condition of the major areas of the plant (e.g., reactor building, auxiliary building, standby cooling tower, and diesel generator building).

The cover letter to the IPE states that the analysis represents the plant configuration as of the IPE submittal date (February 1, 1993), although the official freeze date of the IPE is April 1, 1991 (Section 2.3 of IPE).

E.3 Back-End Analysis

The RBS IPE produces a point estimate of core damage frequency (CDF) equal to 1.55 x 10^{-5} per reactor-year due to internal initiators; the mean CDF evaluated in the uncertainty analysis is equal to 1.87 × 10^{-5}/r-yr (Quantitative results presented in the IPE relate primarily to the point-estimate analysis). Of the total internal-events CDF, 86.3% is due to station blackout (SBO) events, 7.9% is due to transient events, 4.3% is due to loss of offsite power events, and 1.5% is due to transient induced loss of coolant accident (LOCA) events. No ATWS sequences and no Interfacing Systems LOCA (ISLOCA/containment bypass) sequences survived the Level-1 truncation process.

After the submission of the IPE documentation to the NRC, plant modifications were made (see Section E.4) that lead to a calculated decrease in the CDF from 1.55 x 10^{-5} per reactor year to 3.55 x 10^{-6} per reactor year.

The conditional probability of containment failure calculated in the IPE submittal is 0.384. The conditional probability of penetration failure is 0.268 (including 0.028 due to failure to isolate), and the conditional probability of gross (anchor/dome) failure is 0.116. The dominant contribution to radioactive releases were found to be due to SBO sequences and loss of containment isolation sequences. In addition, only SBO sequences were found to lead to large (vessel anchor or dome) containment failure. The total large release frequency calculated by the submittal is 1.84×10^{-6} per reactor year. However, the IPE does not develop a meaningful, consistent definition of "large release."

The Level-2 portion of the IPE submittal contains sufficient information Generic Letter 88-20 and relevant supplements. The licensee has developed a meaningful model of severe-accident progression, has demonstrated a generally adequate appreciation of severe-accident progression, has gained a meaningful quantitative understanding of risks of radioactive releases, and has identified and studied meaningful potential changes to hardware and procedures to mitigate severe accidents.

E.4 Containment Performance Improvement (CPI) Issues

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.

- Improved hydrogen ignitor power supply.

Alternative Water Supply for Drywell/Spray Injection: Alternative water for RPV injection can be provided in the RBS by the diesel driven Fire Protection Water (FPW) pump. The pump can be aligned via the SSW crosstie to the RHR for injection into the vessel. Since this is a low pressure pump, the reactor must be depressurized to allow injection. When DC power is lost as in a long-term SBO, control of SRVs is lost, and FPW is lost as a source injection to the vessel. Thus an alternate source of DC power is needed. A portable DC generator can provide extended DC power during a SBO.

Three plant changes were made after the IPE submittal was completed to realize this alternate water supply:

- A portable DC generator was added to provide DC loads for SBO.

- Internals from three check valves in the FPW/SSW cross-tie were removed, and manual gate valves were added.

- A SBO procedure was modified to provide a different FPW injection path for SBO which uses valves outside the containment.

The impact of these modifications on the CDF was found to reduce the CDF to 3.55×10^{-6} per reactor year. The impact of these modifications on the results of containment analyses and radionuclide releases are calculated, by propagating through the Level-2 analysis. The impact of improved SBO capability is calculated to be the following: 35% reduction in gross containment failure frequency, 43% reduction in the frequency of penetration failure, the conditional probability of containment remaining intact increases from 0.616 to 0.773. These plant modifications were made in 1993 after the IPE documentation was submitted to the NRC. Alternate injection to the drywell was not considered since the RBS plant does not have drywell sprays.

Enhanced reactor pressure vessel depressurization system reliability: The River Bend depressurization system consists of 16 safety relief valves. A Level-1 sensitivity analysis was performed for adding a diesel generator for enhanced reliability of the DC power to SRVs. Loss of instrument air to the SRVs during a station blackout was assessed based on the SRV accumulator size and the addition of two temporary diesel driven compressors. One of the two temporary compressors is to be installed in 1995. Thus, the licensee has addressed the need to enhance the reactor pressure vessel depressurization reliability.

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 RBS EOPs in 1987.

Hardened Vent: RBS currently has a 3-inch (hydrogen purge) vent line. This vent path was determined to be insufficient in averting containment over-pressure failure for events with loss of all containment heat removal. The vent size require to avert such failure was determined to be 10 inches. Hence, the risk impact of installing a 10-inch hardened vent line was evaluated in the IPE.

The level-1 impact of this potential plant change was studied by identifying those existing core-damage sequences that would be recovered as a result of the increased vent capability. Four such sequences were identified that had survived the initial truncation process. Cut sets for these four sequences were deleted from the overall core-damage cut sets listing, and a new estimate of CDF was made, to reflect the improved venting capability. The CDF was found to decrease only slightly, from 1.55×10^{-5} to 1.53×10^{-5}.

From a back-end perspective, the improved venting capability was judged in the IPE to bring about a trade-off in its impact. The beneficial impact of this potential plant change, as evaluated in the submittal, is to essentially eliminate gross containment failure. On the other hand, the increased venting capability was found to lead to large radiological releases (albeit monitored releases). The reduction in the magnitude of radioactive releases is associated with the elimination of gross containment failure is, therefore, suggested in the IPE submittal, to be accompanied by a corresponding increase in the magnitude of radioactive releases due to venting. The IPE implies that increasing hardened vent capability is not warranted. The licensee's analysis of the increased hardened vent capability provides some meaningful insights on the CDF. However, the licensee evaluation of the impact on containment failure and radionuclide releases, is purely hypothetical. The IPE itself acknowledges that radioactive releases through the increased hardened vent would be monitored and could be isolated at some point during the severe accident. Operator action would be involved in isolating the vent. Installation of increased hardened vent capability would probably not eliminate the risk of gross containment failure. Containment failure pressure is not known with sufficiently high certainty to be able to say that a 10-inch vent would completely preclude the possibility of containment failure; but, even if a 10-inch vent path was sure to avert containment failure, the operator may not be able to (or may not) open the vent path. Thus it would have been worthwhile on the part of the licensee to assess the impact of the increased hardened vent capability.

Improved AC Power Supply to the Hydrogen Ignitors: The RBS IPE evaluated the potential modification to the electric supply to the hydrogen ignitors that would help ensure function of the ignitors under SBO conditions. Such modification could possibly reduce the possibility of high containment loads that may otherwise be realized from hydrogen deflagrations and detonations. The impact of this change was found to reduce the conditional probability of penetration failure (expressed as a percentage of the CDF) by 9.7% (from 26.8% to 24.2%) and to reduce the conditional probability of containment structural failure by 12% (from 11.6% to 10.2%). On this basis, the IPE concludes that a hardware upgrade to provide uninterrupted

power supply to the hydrogen ignitors is not warranted. Related to the issue of power supply to hydrogen ignitors, however, the IPE does suggest the need for change to abnormal operating procedure AOP-0050, which applies during SBO events. The revision to this AOP instructs the operators to turn off the igniters if AC power is unavailable.

Although the IPE provides significant insights with respect to impacts of providing alternate power for the hydrogen igniters, we believe this issue is worthy of further analysis. As the IPE itself mentions (Section 6.2.3) and as pointed out by the licensee's reviewers (Sections 5.3 and 5.4), additional evaluation of hydrogen ignitor use under SBO scenarios is required. We recommend that the CPI issue of providing alternate power to igniters be addressed after all issues pertaining to hydrogen ignitor use during SBOs are resolved.

In response to the NRC questions, the licensee stated that increasing the hardened vent capacity and provision of alternate power source for the hydrogen ignitors, was not cost-effective.

E.5 Vulnerabilities and Plant Improvements

The licensee defines "vulnerability" at the plant level, based on comparison of plant CDF with the (licensee-defined) NRC CDF safety goal of 1×10^{-4} per reactor year. On this basis, the licensee concludes that there are no vulnerabilities at RBS.

The IPE submittal states that RBS does not meet a large release safety goal of 1×10^{-6}. In response to NRC questions, the licensee stated that no vulnerability exists based on "the small changes in containment failure for the sensitivity cases performed for the Level 2 IPE". The licensee stated that the large containment failure frequency is driven by the fact that the core damage frequency is primarily due to station blackout. Plant modifications made after the completion of the IPE, have reduced the CDF from 1.55×10^{-5} per reactor year to 3.55×10^{-6} per reactor year. Hence, the licensee apparently concludes that no vulnerabilities exist in the RBS plant.

The licensee has implemented one system modification and three procedural changes, as a result of the Level-1 IPE. From a Level-2 perspective, the following actions appear to be the most beneficial:

- Implementation of plant changes for improved SBO injection capability.
- Implementation of the change to AOP-0050, as described above.

E.6 Observations

The back-end portion of the RBS IPE submittal provides a substantial amount of information in regard to the subject areas identified in Generic Letter 88-20 and NUREG-1335. The 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 III containments. The treatment of phenomenological issues in the CET is quite detailed, and the IPE makes use of results from NUREG-1150 analyses. As an overall observation, the RBS back-end analysis follows closely the methodology and inputs used in NUREG-1150 for Grand Gulf. The major differences between the River Bend and Grand Gulf plants (i.e., those noted in Section 2.2) have been reasonably well accounted for in the RBS Level-2 model. The minor differences (e.g., reactor pedestal pressure capacity) have, in some instances, not been meaningfully modeled. Sensitivity studies have been undertaken by the licensee to assess changes in containment failure probabilities associated with a number of uncertain parameters used as inputs in the CET split fraction quantification process.

The most significant finding of the licensee is the possible reduction in CDF (and in radiouclide release frequency) due to provision of alternate injection and back-up power source in SBO sequences. The impact of these modifications on the CDF was found to reduce the CDF to 3.55 x 10^{-6} per reactor year. The impact of these modifications on the results of containment analyses and radionuclide releases are calculated, by propagating through the Level-2 analysis. The impact of improved SBO capability is calculated to be the following: 35% reduction in gross containment failure frequency, 43% reduction in the frequency of penetration failure, the conditional probability of containment remaining intact increases from 0.616 to 0.773. These modifications (i.e., installation of portable DC generator, removal of internals from three check valves in the FPW/SSW cross-tie, and addition of manual gate valves, and modification of a SBO procedure to provide a different FPW injection path for SBO which uses valves outside the containment) were made after the IPE documentation was submitted to the NRC.

A number of minor weaknesses were identified based on the present review of the submittal, and they include the following:

- The number and variety of deterministic severe accident analyses (i.e., MAAP simulations) appear to be minimal.

- The treatment of ex-vessel steam explosion in the reactor cavity is weak. The water-tight door cannot just be assumed to have the same capacity as the pedestal (whose capacity is not calculated); there is some probability that the door will not keep water out of the pedestal cavity, and thereby the potential for ex-vessel steam explosions cannot be ruled out.

- The analysis models basemat melt-through, but completely eliminates its possibility in quantification.

- There is minimal or no treatment of operator actions in back-end analyses.

- The meaning of the IPE's definition of large release is somewhat arbitrary. Meaningful summaries (i.e., not just the 59 STC frequencies) pertaining to the timing and magnitude of radioactive releases are not reported.

- When split fractions are developed for the DET/CET quantification based on engineering judgment, the rationale for the quantification (or the choice) of the split fractions are often not presented.

The important points of the technical evaluation of the River Bend IPE back-end analysis are summarized as follows:

- The back-end portion of this IPE submittal, for the most part, is relatively well performed, and provides a substantial amount of information in regard to the subject areas identified in Generic Letter 88-20 and NUREG-1335.

- The submittal includes most phenomena of interest to severe accident phenomenology for BWRs with Mark III containments; the level of CET modeling detail is apparently comparable to that for the NUREG-1150 analysis of Grand Gulf..

- The treatment of phenomenological issues in the CET is mostly reasonable, and the IPE makes appropriate use of results from NUREG-1150 analyses, and from other NRC-sponsored research.

- The licensee evaluated meaningful plant improvements, and implemented one group of plant modifications, which lead to calculated reductions in CDF, frequencies of containment failure and radionuclide release.

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NOMENCLATURE

AC	Alternating Current
ADS	Automatic Depressurization System
AOP	Abnormal Operating Procedure
AOV	Air Operated Valve
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
CIS	Containment Isolation System
CPI	Containment Performance Improvement
CRVICS	Containment and Reactor Vessel Isolation Control System
CS	Containment Spray
CSS	Core Spray System
CST	Condensate Storage Tank
CVS	Containment Venting System
DC	Direct Current
DCH	Direct Containment Heating
DET	Decomposition Event Tree
DF	Decontamination Factor
DG	Diesel Generator
ECCS	Emergency Core Cooling Systems
EOP	Emergency Operating Procedure
EPRI	Electric Power Research Institute
EPS	Electric Power System
ESF	Engineered Safety Features
ESFAS	Engineered Safety Features Actuation System
EVSE	Ex-Vessel Steam Explosion
FMEA	Failure Modes and Effects Analysis
FPS	Fire Protection System
FPW	Fire Protection Water
FRVS	Filtration, Ventilation and Recirculation System
GE	General Electric
GI	Generic Safety Issue
GL	Generic Letter
GSU	Gulf States Utilities
HEP	Human Error Probability
HGE	Hydrogen Generating Event
HPCS	High Pressure Core Spray

NOMENCLATURE (Continued)

HPME	High Pressure Melt Ejection
HRA	Human Reliability Analysis
HVAC	Heating, Ventilation, and Air Conditioning
IAS	Instrument Air System
IN	Information Notice
IPE	Individual Plant Examination
IPEEE	Individual Plant Examination of External Events
ISLOCA	Interfacing Systems Loss of Coolant Accident
IVSE	In-Vessel Steam Explosion
LOCA	Loss of Coolant Accident
LOSP	Loss of Offsite Power
LPCI	Low Pressure Coolant Injection
LPCS	Low Pressure Core Spray
LT-SBO	Long Term Station Blackout
MAAP	Modular Accident Analysis Program
MCC	Motor Control Center
MOV	Motor Operated Valve
MSIV	Main Steam Isolation Valve
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
NSW	Normal Service Water
PCS	Power Conversion System
PDLD	Plant Damage Logic Diagram
PDS	Plant Damage State
PRA	Probabilistic Risk Assessment
RBS	River Bend Station
RCIC	Reactor Core Isolation System
RCS	Reactor Coolant System
RFP	Reactor Feedwater Pump
RHR	Residual Heat Rejection
RPCCW	Reactor Plant Component Cooling Water
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RSS	Reactor Safety Study
SAIC	Science Application International Corporation
SACS	Station Auxiliary Cooling System
SAS	Service Air System
SBO	Station Black-Out
SCT	Standby Cooling Tower
SGTS	Standby Gas Treatment System
SLC	Standby Liquid Control
SLCS	Standby Liquid Control System

NOMENCLATURE (Continued)

SOP	System Operating Procedure
SORV	Stuck-Open Relief Valve
SPC	Suppression Pool Cooling
SPMU	Suppression Pool Make-Up
SRV	Safety Relief Valve
SSW	Station Service Water
STCLD	Source Term Category Logic Diagram
STLD	Source Term Logic Diagram
TER	Technical Evaluation Report
THERP	Technique for Human Reliability Error Rate Prediction
TPCCW	Turbine Plant Component Cooling Water
TW	Loss of Decay Heat Removal
USAR	Updated Safety Analysis Report
USI	Unresolved Safety Issue

1. INTRODUCTION

This Technical Evaluation Report (TER) documents the results of the "submittal-only" review of the River Bend 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 RBS 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 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

RBS is a single-unit, GE BWR/6 with Mark III free-standing steel containment with a reinforced-concrete shield building. The cylindrical shell of the primary containment vessel is anchored 5 feet within the 10-ft thick reactor concrete mat foundation. The vessel has a flat-bottom steel liner that is welded to embedded steel members in the mat foundation. The

secondary containment at RBS consists of the reinforced-concrete shield building, the fuel building, and the auxiliary building. The shield building forms an annular space of 357,400 ft^3 around the containment vessel. A standby gas treatment system (SGTS) keeps the secondary containment at negative pressure, and provides for clean-up of potentially contaminated air volumes during accident conditions. A number of significant safety related components are housed within secondary containment. In addition to the foregoing elements, RBS has containment systems which provide for containment heat removal, containment isolation, combustible gas control, and containment venting.

RBS has a power rating of 2894 MWt. The plant site is located near Baton Rouge, Louisiana. The reactor systems and engineered safeguard systems at River Bend are essentially the same as for other BWR 6/Mark III designs. The primary differences between RBS and Grand Gulf (NUREG-1150 plant) designs and analyses pertain to the following: containment design, containment cooling, containment vacuum breakers, suppression pool make-up, shield building/containment annulus, reactor pedestal cavity, hydrogen ignitor power supplies, and containment venting capability.

The unique containment design characteristics and features at RBS include the following:

- There is an annular region between the free-standing containment vessel and the outer shield building, which provides for some additional decontamination of radionuclides before release to the environment. The air volume in this annular region is filtered by the Standby Gas Treatment System (SGTS).

- Unlike other plants in the U.S. with Mark III containments, where containment sprays and drywell sprays are provided for containment cooling, RBS employs fan coolers. The RBS IPE states that this means of containment cooling helps ensure that the containment remains inerted under post-accident conditions in the containment environment. In addition, fan coolers function independent of the RHR system.

- Unlike other Mark III containments, RBS does not have vacuum breakers or check valves between the containment and drywell atmospheres. The RBS IPE states that this reduces the potential for suppression pool bypass, and that the drywell has sufficient strength to resist possible negative pressure loadings.

- Unlike other Mark III designs, RBS does not provide for suppression pool make-up via upper fuel-pool dump. The IPE submittal states that River Bend maintains sufficient water in the suppression pool, such that make-up is not required. Because the upper pool is always maintained, the drywell head is always submerged, preserving drywell head seal integrity, and thus eliminating the concern of high-temperature seal failure. In addition, the IPE states that the potential for energetic FCIs is reduced due to elimination of accumulation of water in the pedestal cavity that might accompany pool make-up.

- The access door to the reactor pedestal at RBS is water-tight, and the door is kept closed during plant operation. The IPE takes credit for this feature, as means for reducing the potential for energetic FCIs.

- The containment and drywell hydrogen igniters are powered from separate electrical circuits, providing for independent and preferential activation in these two regions. This feature is not modeled in the Level-2 analysis.

- RBS has a 3-inch containment vent line, which is insufficient to prevent containment failure due to over-pressure. Other Mark III containments have sufficient vent capability to reduce the potential for over-pressure failure.

- RBS has a 30-inch containment dome vent which was permanently sealed (welded) after construction. In the IPE, this condition is stated to lead to local failure/leakage (under high-pressure containment vessel loads) that acts essentially as an uncontrolled vent path which reduces the possibility for catastrophic containment failure.

Most of these unique features are modeled specifically in the RBS IPE back-end analysis.

2. TECHNICAL REVIEW

The present review compared the River Bend 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 licensee's IPE process for the back-end evaluation of RBS is stated to be a Level-2 PRA that uses inputs and procedures similar to those employed in the NUREG-1150 [4] analysis of Grand Gulf. The Level-2 evaluation starts with the results of the Level-1 analysis; i.e., a description of possible accident sequences leading to core damage, together with estimates of their annual frequencies of occurrence. A plant-damage state logic diagram is constructed and used to bin core-damage sequences having similar cut sets into given plant damage state (PDS) bins. Forty-three (43) unique plant damage states having non-zero frequencies are obtained in the IPE. A containment event tree (CET) is then constructed and quantified to model accident progression. The CET is supported by decomposition event trees (DETs) for complete modeling and quantification of accident progression. The overall approach is similar to the use of accident progression event trees (APETs) in NUREG-1150, and the results of this part of the analysis produces accident progression bins (APBs) and their annual frequencies of occurrence. The River Bend IPE produced 78 APBs having non-zero frequencies. A Source Term Logic Diagram (STLD) is then used to bin the APBs to Source Term Categories (STCs). The source term analysis results in definition of source term categories and their annual frequencies of occurrence, which represent the end products of the Level 2 analysis. A total of 59 STCs were developed and quantified in the analysis. The licensee has stated that the level of detail employed in the RBS IPE is somewhat greater than that for the NUREG-1150 analysis of Grand Gulf.

The licensee has performed a number of sensitivity studies for the back-end evaluation. The sensitivity studies have examined the impact on containment failure modes. In addition, a meaningful uncertainty analysis was apparently performed, although the details are not fully documented in the IPE. (It is not clear to this reviewer whether this uncertainty analysis was performed for only the Level-1 IPE, or whether Level-2 was also included).

The point estimate CDF obtained from the Level-1 analysis is equal to 1.55×10^{-5}/ry. This point estimate was used as the basis for quantifying plant damage state (PDS) frequencies and other results in the Level-2 analysis. The RBS IPE also estimates that the CDF due to internal flooding equal to 1.75×10^{-8}, and hence, is negligible in comparison to the risk from other internal initiators.

Of the total internal events CDF, 86.3% is due to station blackout (SBO) events, 7.9% is due to transients, 4.3% is due to loss of offsite power events, and 1.5% is due to transient-induced Loss of Coolant Accident (LOCA) events. No ATWS sequences and no Interfacing Systems LOCA (ISLOCA) sequences survived the licensee's Level-1 truncation process.

After the submission of the IPE documentation to the NRC, plant modifications were made (see Section 2.4.2) that lead to a calculated decrease in the CDF from 1.55 x 10^{-5} per reactor year to 3.55 x 10^{-6} per reactor year.

2.1.2 Multi-Unit Effects and As-Built/As-Operated Status

RBS is a single-unit nuclear power plant. The cover letter to the IPE submittal states that the analysis represents the plant configuration at River Bend as of the IPE submittal date (February 1, 1993). The official freeze date of the IPE, however, was April 1, 1991 (see Section 2.3 of IPE).

The latest data on plant design and configuration, technical specifications, operating procedures, maintenance procedures, etc., were stated to be used in the River Bend IPE. In addition, plant walkdowns of the relevant plant locations (reactor building, auxiliary building, standby cooling tower, diesel generator building) were conducted, and videodisc-based surrogate tours were performed, to support the Level-1 and Level-2 IPEs. (The IPE provides only a brief description of the plant walkdown effort, but the description seems to indicate an adequate effort was undertaken and documented so as to define the as-operated condition of the important areas of the plant). Personnel with day-to-day familiarity with the plant had extensive involvement in the IPE.

2.1.3 Licensee Participation and Peer Review of IPE

The back-end analyses of the River Bend IPE was performed as a utility-contractor effort, with the project team consisting of GSU and Halliburton-NUS personnel. EQE Engineering performing the containment fragility analysis. An independent review of the back-end analysis was conducted as follows: Science Application International Corporation (SAIC) reviewed all Level 2 analyses; and Stone and Webster Engineering Corporation reviewed the containment fragility analysis. In addition, Enercon Services provided comments on Level 2 IPE assumptions concerning hydrogen generation.

GSU clearly had a major participation in the oversight and execution of the IPE, although this involvement may have been somewhat less in the Level-2 analysis than in the Level-1 analysis. NUS clearly had a substantial involvement in technical aspects of the Level-2 analyses. The IPE was performed on-site at River Bend, and consultants visited the site whenever plant-specific information concerning the layout of components, maintenance practices, or specific operational information was required. As stated above, GSU staff involved in the IPE included those having day-to-day familiarity with the plant. The comments of the reviewers are documented in Section 5 of the IPE. Although each comment was addressed on a case-by-case basis by GSU, as

summarized in Section 5 of the IPE, the impact of a number of commer[illegible]ave not yet been included in the IPE model and results.

Licensee involvement and peer review appear to be adequate with respect to the objectives of GL 88-20 and the related procedural and information guidelines of NUREG-1335.

2.2 Containment Analysis

The key characteristics of the River Bend plant and containment were identified in Section 1.2 of this review. The River Bend plant is very similar to the Grand Gulf plant. The significant containment features at River Bend are described in Section 4.1, including Tables 4.1-1 and 4.1-2 and Figures 4.1-1 through 4.1-13 of the submittal.

Table 1 provides a comparison of containment features between the River Bend plant and the Grand Gulf plant, another BWR 6 plant with a Mark-III containment. It can be seen that the containment features are comparable between the two plants. In addition, those plant-specific features listed under Section 1.2 are important for accident progression in the River Bend plant.

Table 1 shows that the ratios of containment free volume-to-power and the suppression water volume-to-power are somewhat larger for River Bend than for Grand Gulf. This can have a small impact on the containment failure probability calculated by the IPE submittal.

Table 1 Comparison of River Bend and Grand Gulf Plant and Containment Design Features that Contribute to The Progression of Severe Accidents

Feature	River Bend	Grand Gulf
Rated Power, MWt	2,894	3,833
Core UO_2, kg	125,194	165,704
Containment Free Volume, m^3	34,000	39,640
Suppression Pool Volume, m^3	3,625	3,851
Drywell Free Volume, m^3	7,100	7,650
Pedestal Free Volume, ft^3	178	268
S. Pool Water Vol./Power, m^3/MWt	1.25	1.0
Containment Vol./Power, ft^3/MWt	11.7	10.3
Containment Construction	Free Standing Steel with Concrete Shield Building	Steel Lined, Reinforced Concrete
Failure Pressure	63 psig	67 psig

2.2.1 Front-End/Back-End Dependencies

The results of the front-end event trees in the RBS IPE are accident sequences and their frequencies. A small event tree/large-linked fault tree approach was employed in the front-end analysis. For the Level-1/Level-2 interface analysis, core damage cutsets from the Level-1 analysis are grouped into similar Plant Damage States (PDSs). Each PDS, therefore, is an aggregation of Level-1 cutsets that are expected to have similar severe-accident progressions. Three steps were employed in grouping cutsets into PDSs: (1) cutset analysis after core damage, involving determination of plant conditions and equipment availabilities; (2) system analysis after core damage, identifying containment and ECCS information not included in Level 1, but required for Level 2; and (3) PDS binning analysis, using a Plant Damage Grouping Logic Diagram (PDLD).

Cutset Analysis After Core Damage: Each cutset from Level-1 was examined, and information required for CET analysis was identified and coded. The sequence, initiating event, and cutset were used as inputs to this process. Section 4.3.1.1 of the IPE describes the PDS designator coding scheme used in the IPE to describe the cutset and sequence information required for subsequent analysis. Eight attributes (each designated by a two or three character field) were developed for this coding: (1) initiating event type (18 categories); (2) availability of power at time of core damage (3 categories); (3) status of HPCS (3 categories); (4) status of low-pressure injection, LPCI and LPCS (3 categories); (5) status of suppression pool cooling (SPC) system (3 categories); (6) status of containment fan cooling (CFC) system (3 categories); (7) reactor pressure at time of core damage (2 categories); and (8) time of injection failure, in hours, after the initiating event.

For the Level-1 analysis, a truncation limit of 1×10^{-9} was applied to core-damage sequences passing through to the Level-2 analysis. No LOCA, ATWS or V-sequence cutsets survived this truncation; only Station Blackout (SBO) and transient events were passed to the Level-2 analysis. Thus, only eleven (11) of the eighteen (18) total initiating event were actually involved in the Level-1/Level-2 interface analysis. Nonetheless, the Level-1/Level-2 interface model for River Bend is capable of analyzing all sequences should they be identified as quantitatively significant in the future. Section 4.3.1.2 of the IPE describes the assumptions made in determining the applicable category for a given attribute.

The results of this phase of the interface analysis are PDS attribute codes and frequencies for each combination of attributes that are relevant to the plant. Forty-four (44) attribute combinations were produced in this assessment.

System Analysis After Core Damage: The inputs to this second phase of the Level-1/Level-2 interface analysis were the PDS attribute codes and attribute combination frequencies described above. The result of this phase was information concerning containment and ECCS not evaluated in the Level-1 analysis, but required for Level-2.

Six questions were posed to define the availability of six systems. These questions help to determine the status of those attributes mentioned above that were initially assigned a status of "unknown" based on the Level-1 analysis results only. Following are the six questions:

1. Containment isolation: success (P=0.9674) or failure (P=0.0326).

2. Containment venting: success (P=0.9) or failure (P=0.1).

3. Power recovery: early enough to prevent vessel failure, early enough to prevent containment failure, or failure to recover power.

4. Emergency injection: recovery of high-pressure injection (P=0.951), recovery of low-pressure injection (P=0.995), or no late injection recovery.

5. Suppression pool cooling: success (P=0.996) or failure (0.004).

6. Containment fan cooling: success (P=0.998) or failure (0.002).

The answers to these six questions were appended to the preceding PDS designator codes. Then a truncation frequency of 1×10^{-10} was applied. The analysis produced 263 appended PDS group codes and their frequencies.

PDS Binning Analysis Using PDLD: The Plant Damage Grouping Logic Diagram (PDLD) used in the final phase of the Level-1/Level-2 interface analysis is shown in Figure 4.3-1 of the IPE. This logic diagram has 12 event nodes, as described below:

1. PDS:CNTBYP - "Not a Containment Bypass Sequence": Because containment bypass provides a direct release path to the environment, such sequences are grouped into a separate PDS. No such sequence, however, survived the Level-1 truncation process; the RBS IPE model is, nonetheless, capable of evaluating such sequences.

2. PDS:CNTFAL - "Containment Intact at Core Damage": This node addresses whether or not the containment is intact at the time of core damage. Containment failure prior to core damage results in larger radiological releases to the environment than a later failure, since fission products can be released prior to decontamination within an intact containment.

3. PDS:EVENT1 - "Event Type: Containment Intact": Sequences that do not involve early containment failure are further divided into the following three categories: (1) SBOs, (2) LOCAs, and (3) others. SBOs are grouped separately due to unique hydrogen explosions concerns for such sequences. LOCAs are grouped separately because they can bypass the suppression pool. All other sequences are treated similarly in the subsequent analysis.

4. PDS:EVENT2 - "Event Type: Containment Not Intact": Sequences that do involve early containment failure are further divided into the following two categories: (1) ATWS sequences involving early containment failure, and (2) other sequences with containment heat removal failed but injection initially available. ATWS sequences are grouped separately because they can overload the suppression pool.

5. PDS: CNTISOL - "Containment Isolated at Vessel Failure": This node addresses containment isolation status, which can be one of the following: (1) isolated or (2) not isolated. If the containment is not isolated, there will be an early and continuous release of fission products.

6. PDS: TINJFAIL - "Vessel Injection Failure Time": This node is important for SBO sequences. The length of injection time impacts the amount of steam produced in the containment which, in turn, impacts the possibility for subsequent hydrogen explosion. The possible branches for this node include: (1) injection failure occurs within the first two hours; (2) injection fails between 2 to 4 hours; and (3) injection failure occurs after 4 hours.

7. PDS:PWRREC - "Offsite Power Recovery Time": This node has major significance for SBO sequences. Recovery of offsite power may lead to hydrogen explosion in the containment, particularly if the igniters are left on and are functioning at the time of power recovery. The possible branches from this node are: (1) recovery prior to reactor vessel failure; (2) recovery prior to (early) containment failure; and (3) no power recovery.

8. PDS:CHR - "Containment Heat Removal": This node addresses whether or not containment heat removal is functioning via suppression pool cooling or fan cooler units. Hence, sequences are separated into those associated with containment heat removal, and those not. The two possible branches are: (1) CHR available and (2) no CHR available.

9. PDS:VENT - "Containment Vent": Late containment over-pressure can be controlled via venting. The 3-inch vent path at RBS cannot avert rapid over-pressure failure, such as from hydrogen burns. The two possible branches for this node are: (1) containment venting available and (2) no containment vent available.

10. PDS:LATINJ - "Late In-Vessel Injection": The existence and mode of late vessel injection (i.e., after vessel failure) impacts the ability to cool the core debris in-vessel and ex-vessel. There are three possible branches for this node: (1) high pressure injection is available, or both high pressure and low pressure injection systems are available; (2) low pressure injection only is available; and (3) no late injection is available.

11. PDS:RXPRES - "RPV Depressurization at Vessel Failure": The mode in which core debris is expelled from the reactor vessel will impact how the method of debris dispersal in the pedestal and remaining drywell, thus impacting the coolability of debris. The possible branches are: (1) RPV depressurized at vessel breach, and (2) RPV not depressurized before vessel breach.

12. PDS:OLDPDS - "Convert Old PDS Number to New PDS Number": This node simply converts to a new sequence of PDSs, numbering 1 to 43. A total of 141 plant damage states are initially identified as a results of application of this PDLD to the appended PDS designators. Only 43 of these have non-zero frequencies.

PDS Binning Results: The results of the Level-1/Level-2 interface analysis are the 43 plant damage states (that survive the 10^{-10} truncation limit applied during the second step of the interface analysis) and their frequencies. Based on application of the PDLD to the earlier PDS designator attributes, the plant damage state groups convey the information significant to Level-2 analysis. The input to the Level-2 analysis is, therefore, the 43 PDS categories and their frequencies, and Level-2 significant characterization of each PDS.

A review of the interface analysis suggests that it is a reasonable and valid approach. However, the analysis is not documented in complete detail. It would have been somewhat more consistent to wait to apply the 1×10^{-10} truncation frequency until the end of the entire PDS binning process. This was probably not done, in order to simplify the effort during the third phase of the interface analysis. This truncation approach, nevertheless, is not viewed as a limitation in the analysis or its results.

2.2.2 Containment Event Tree Development

Probabilistic modeling and quantification of severe accident progression for the probabilistically significant plant-damage bins was performed using containment event trees (CETs) and Decomposition Event Trees (DETs, i.e., sub-event trees). The CET analysis starts with the 43 PDSs (and their frequencies) evaluated from the Level-1/Level-2 interface analysis. The result of the CET/DET quantification process are accident progression bins (APBs) and their estimated frequencies of occurrence. The main CET developed for River Bend has thirteen top event nodes, each of which is further developed using the DET subtrees. The CET quantification, in general, relies heavily on data developed for CET input quantification in the NUREG-1150 analysis of Grand Gulf. In a limited number of cases, the CET quantification is performed using results from plant-specific MAAP analyses.

Figure 4.5-1 of the submittal depicts a single RBS CET, which is apparently applicable for all types of initiators (station blackout, LOCAs, transients, transients with loss of decay heat removal, and ATWS sequences) and plant damage states. Thus, the same CET is used for all PDSs. The top events modeled, together with a description of their supporting DETs, are summarized and discussed below.

Event 1, CET:INV-COOL: "Debris Cooled In-Vessel"

This node determines the relative probability that the core debris can be cooled in-vessel and reactor vessel failure may be averted. The two possible outcomes of this node are: (1) debris is cooled in-vessel (vessel failure prevented); or (2) debris is not cooled in-vessel (vessel failure occurs). The DET (subtree) for this node consists of the following five DET top events:

1. DET(INV_COOL) - LONG_HR "Long-Term Heat Removal from Containment": The availability of long-term containment heat removal is quantified directly from PDS: CHR. The possible branches for this sub-node are: (1) containment heat removal is successful; (2) containment heat removal is not successful.

2. DET(INV_COOL) - LATE_INJ "Status of Late In-Vessel Injection": The availability of late injection is quantified directly from PDS:LATINJ. The possible branches for this sub-node are: (1) HP injection; (2) LP injection; (3) no late water injection; or (4) late water injection, but the reactor is not shut down.

3. DET(INV_COOL) - RXPRES "Status of Reactor Vessel Pressure": The status of reactor vessel pressure is quantified directly from PDS:RXPRES. The possible branches for this sub-node are: (1) RPV is depressurized prior to core plate failure; or (2) RPV is not depressurized during core damage.

4. DET(INV_COOL) - MOLTEN_VB "Fraction of Molten Material in the Vessel at the Time of Vessel Breach": This branch is quantified using a split fraction taken directly from NUREG/CR-4551 [5] for Grand Gulf. The possible branches for this sub-node are: (1) large mass (~40% of core mass) of molten debris in lower reactor vessel (P=0.025); or (2) small mass (~10% of core mass) of molten debris in lower reactor vessel (P=0.975).

5. DET(INV_COOL) - INV_COOL "Debris Cooled In Vessel": This branch is quantified using a split fraction taken directly from NUREG/CR-4551 for Grand Gulf. The particular split fraction applied to a given branch depends on the path taken in the DET analysis. The possible branches for this sub-node are: (1) debris is cooled in vessel (P=0.50 (large MOLTEN_VB); P=0.50 (small MOLTEN_VB)); or (2) debris is not cooled in vessel (P=0.75 (large MOLTEN_VB); P=0.25 (small MOLTEN_VB)).

Event 2, CET:V_ERLY_CF: "Mode of Very Early Containment Failure"

This node determines the conditional probability of each potential containment failure mode that may occur as a result of hydrogen burn or explosion. The four possible outcomes of this node are: (1) gross failure; (2) penetration failure; (3) no failure, no hydrogen burn occurs; or (4) no failure, hydrogen burn does occur. The DET (subtree) for this node consists of the following ten DET top events:

1. DET(V_ERLY_CF) - SBO "Station Blackout Status": This branch in the DET is quantified directly from PDS:EVENT1. The possible branches for this sub-node are: (1) SBO or (2) no SBO.

2. DET(V_ERLY_CF) - CONT_ISOL "Containment Isolation Status": This branch in the DET is quantified directly from PDS:CNTISOL. The possible branches for this sub-node are: (1) containment successfully isolated; or (2) containment not isolated.

3. DET(V_ERLY_CF) - STM_CONC "Containment Steam Concentration": The amount of steam concentration is used to determine whether or not hydrogen burn can occur. This branch in the DET is quantified by means of MAAP calculations, with input on injection failure time coming directly from PDS:TINJFAIL. The possible branches for this sub-node are: (1) 0-35% containment steam concentration; (2) 35-55% containment steam concentration; or (3) greater than 55% containment steam concentration.

4. DET(V_ERLY_CF) - H2_INVESS "Fraction of Zircaloy Inventory Reacted In-Vessel": This branch is quantified using split fractions taken from NUREG/CR-4551 for Grand Gulf, and modified using engineering judgment. MAAP calculations are used to evaluate the amount of zirconium reacted in-vessel for each important SBO type scenario. The hydrogen concentration in containment (next branch in DET) is evaluated from the fraction of zirconium reacted and the steam concentration. The possible branches for this sub-node are: (1) less than or equal to 10% in-vessel zircaloy is reacted (P=0.05); (2) between 10% and 20% of in-vessel zircaloy is reacted (P=0.20); or (3) greater than or equal to 30% in-vessel zircaloy is reacted (P=0.75).

5. DET(V_ERLY_CF) - H2CONC "Containment Hydrogen Concentration": This DET node has a single branch characterized by a value that is determined from the previous DET outcomes. The possible values for this branch are: (1) less than 10% containment hydrogen concentration; (2) 18% containment hydrogen concentration; or (3) 25% containment hydrogen concentration.

6. DET(V_ERLY_CF) - H2_I_REC "Ignition from Recovery of Hydrogen Igniters": This branch in the DET is quanitified from PDS:PWRREC (to determine time of power recovery for SBO events) and PDS:EVENT1 (to determine the type of event). The possible branches for this sub-node are: (1) ignition occurs from recovered igniters; or (2) no ignition from hydrogen igniters.

   With respect to evaluation of this sub-node, it is important to point out that a human error analysis was conducted to evaluate the probability that the hydrogen igniters are not turned on, and then off, correctly. The human error probability (HEP) that the igniters are in the "on" condition when power is recovered was evaluated to be 0.07. This is the only occurrence of reference to human error analysis in the back-end portion of the IPE. Yet, this HEP result was apparently not even used in quantifying the DET/CET model (see pg. 607 of IPE).

7. DET(V_ERLY_CF) - RAN_IGNRN "Ignition from a Random Spark Occurs": This branch in the DET is quantified using engineering judgment. The possible branches for this sub-node are: (1) ignition (P=0.50); or (2) no ignition (P=0.50). (The split fraction values of P=0.50 apparently indicate that the IPE team was uncertain with respect to this assignment).

8. DET(V_ERLY_CF) - SMALL_BURN "Ignition While Hydrogen Concentration Is Low, Resulting in Benign Burn": This branch in the DET is quantified using engineering judgment in consideration of NUREG/CR-4551 results. The possible branches for this sub-node are: (1) yes, small burn occurs; or (2) no, no small burn occurs. The values for split fractions assigned depend on steam concentration, zircaloy reacted in-vessel, and the outcome of ignition due to recovered hydrogen igniters. Twelve cases are identified on p. 609 of the IPE.

9. DET(V_ERLY_CF) - BURN_TYPE "Detonation or Large Burn": This branch in the DET determines the most likely type of hydrogen burn. This branch is quantified using engineering judgment in consideration of NUREG/CR-4551 results. The possible branches for this sub-node are: (1) detonation; or (2) large burn. The values for split fractions assigned depend on steam and hydrogen concentration levels in the containment.

10. DET(V_ERLY_CF) - V_ERLY_CF "Mode of Early Containment Failure": This branch in the DET sorts sequences into the potential containment failure modes, based on results of the containment failure analysis (described in next subsection). Two MAAP calculations are used to evaluate peak pressures based on conditions of hydrogen deflagration or detonation within the containment. The possible branches for this sub-node are: (1) gross failure; (2) penetration failure; (3) no failure, no burn; or (4) no failure, no burn.

    The two MAAP cases do not result in meaningful differences in peak pressure; furthermore, temperature conditions within the containment are not described. The number and description of MAAP runs for this case are thought to be inadequate. The same can be said concerning the similar DET analyses for early containment failure and late containment failure.

Event 3, CET:INJ_FAL1: "Very Early Containment Failure Fails All In-Vessel Injection"

This node evaluates the effects of containment failure on the continued availability of injection system (pumps and valves) between the time core damage occurs and vessel failure occurs. The two possible outcomes of this node are: (1) no injection failure; or (2) injection failed. The DET (subtree) for this node consists of the following three DET top events:

1. DET(INJ_FAL1) - CF_MODE "Containment Failure Mode": This branch sorts out the very early containment failure modes (gross failure or penetration failure). It is

quantified based on node CET:V_ERLY_CF. The possible branches for this sub-node are: (1) gross failure; or (2) penetration failure.

2. DET(INJ_FAL1) - PIPE_FAIL "Injection Piping Not Disrupted (from Anchor Dome Failure)": This branch in the DET determines if injection piping fails due to gross containment failure or penetration failure. This question was not included in NUREG/CR-4551 results. Failed injection piping may eject hot gases that disable critical equipment. The possible branches for this sub-node are: (1) piping not disrupted (P=1.0, penetration failure; P=0.1, gross failure); or (2) piping disrupted and injection failed (P=0.0, penetration failure; P=0.9, gross failure). The values for split fractions assigned are based on engineering judgment.

3. DET(INJ_FAL1) - INJ_FAL1 "Injection Power Supplies/Motor/Pumps Not Failed": This branch defines the result from this DET. It is quantified based on engineering judgment. The possible branches for this sub-node are: (1) pumps and power supply available (P=0.5); or (2) not available (P=0.5) A sensitivity analysis is conducted in the IPE for these split fractions. It is not clear from the IPE documentation and licensee responses to the NRC questions [10] how the split fractions for this node were derived.

Event 4, CET:EARLY-DW: "Drywell Fails At/Near RV Failure"

This node evaluates the probability of drywell failure occurring at the time of reactor vessel failure. This can lead to the bypass of suppression pool scrubbing of fission products. The two possible outcomes of this node are: (1) drywell failure at RPV failure; or (2) no drywell failure at RPV failure. The DET (subtree) for this node consists of the following six DET top events:

1. DET (EARLY_DW) - RV_PRESS "RPV Pressure at RPV Failure": This sub-node is quantified directly from PDS:RXPRES. The possible branches for this sub-node are: (1) RPV is at high pressure (i.e., $\geq$ 200 psig); or (2) RPV is at low pressure (i.e., $<$ 200 psig).

2. DET(EARLY_DW) - VF_SIZE "Size of RV Failure": This sub-node determines the probability of the size of the RPV failure, which impacts the rate of blowdown of the vessel. Results from NUREG/CR-4551 for Grand Gulf are used to quantify the split fraction for breach size. The two possible branches for the sub-node are: (1) small breach (0.1 m^2) (P=0.75); or (2) large breach (2.0 m^2) (P=0.25).

3. DET(EARLY_DW) - PED_OP "Pedestal Fails Due to Over-Pressure": This sub-node evaluates if the reactor pedestal fails due to internal over-pressure. Split fractions and pedestal capacity results from NUREG/CR-4551 for Grand Gulf are used to quantify this sub-node. The two possible branches for this sub-node are: (1) pedestal over-pressure failure; or (2) no pedestal over-pressure failure. The split fraction values depend on the reactor vessel pressure and the breach size (p. 616 of IPE).

It is important to point out here that the reactor pedestal design is significantly different for RBS, as compared to that for Grand Gulf. Not only is the pedestal size different, but it is likely that the arrangement of cut-outs in the pedestal (which can have a major effect on pedestal structural capacity) is also not the same. The licensee review made a mention of this concern.

4. DET(EARLY_DW) - PED_DW_F "Pedestal Failure Causes Drywell Failure": This sub-node evaluates if the reactor pedestal failure causes major failure of the drywell. Split fractions for this sub-event are quantified using expert judgment in the IPE. The two possible branches for this sub-node are: (1) pedestal failure fails drywell (P=0.125); or (2) pedestal failure does not fail drywell (P=0.875).

5. DET(EARLY_DW) - DW_OP_F "Drywell Overpressure Failure at RV Failure": Another mode of drywell failure is direct over-pressurization. Split fractions for this node are based on expert judgment. The two possible branches for this sub-node are: (1) drywell over-pressure failure (P=0.05); or (2) no drywell over-pressure failure (P=0.95).

6. DET(EARLY_DW) - EARLY_DW "Drywell Fails At/Near RV Failure": This sub-node reflects the outcome of this DET.

Event 5, CET:EARLY-CF: "Mode of Early Containment Failure"

This node determines the conditional probability of containment failure due to each potential failure mode. The four possible outcomes of this node are: (1) gross failure; (2) penetration failure; (3) no failure, no hydrogen burn occurs; or (4) no failure, hydrogen burn does occur. The DET (subtree) for this node consists of the following nine DET top events:

1. DET(EARLY_CF) - SBO "Is this PDS for an SBO Event": This branch in the DET is quantified directly from PDS:EVENT1. The possible branches for this sub-node are: (1) SBO, station blackout sequence; or (2) no SBO sequence.

2. DET(EARLY_CF) - IGN_REC "Were Igniters Recovered in DET:V_ERLY_CF": This branch in the DET sorts the PDS into two groups based on previous recovery status of hydrogen igniters. The DET node is quantified directly by PDS:PWRREC and PDS:EVENT1. The possible branches for this sub-node are: (1) ignition from recovered recombiners; or (2) no ignition from recombiners.

3. DET(EARLY_CF) - PREV_BURN "Did Hydrogen Burn Occur In DET:V_ERLY_CF": This is an information node only, with results carried over from previous nodes. The two possible branch outcomes are: (1) previous burn occurred; or (2) no previous burn occurred.

4. DET(EARLY_CF) - DW_FAIL "Previous Drywell Failure": This branch in the DET sorts the PDS into two groups based on previous occurrence of drywell failure. The DET node is quantified directly from CET:EARLY_DW. The possible branches for this sub-node are: (1) previous drywell failure occurred, or (2) no previous drywell failure occurred.

5. DET(EARLY_CF) - STM_CONC "Containment Steam Concentration": The meaning and quantification of this node are the same as those developed previously for DET(V_ERLY_CF):STM_CONC.

6. DET(EARLY_CF) - H2_INVESS "Fraction of Zircaloy Inventory Reacted In-Vessel": The meaning and quantification of this node are the same as those developed previously for DET(V_ERLY_CF):H2_INVESS.

7. DET(EARLY_CF) - H2CONC "Containment Hydrogen Concentration": The meaning and quantification of this node are the same as those developed previously for DET(V_ERLY_CF):H2CONC.

8. DET(EARLY_CF) - BURN_TYPE "Detonation or Large Burn": The meaning and quantification of this node are the same as those developed previously for DET(V_ERLY_CF):BURN_TYPE.

9. DET(EARLY_CF) - EARLY_CF Mode of Early Containment Failure": The meaning and quantification of this node are the same as those developed previously for DET(V_ERLY_CF):V_ERLY_CF.

   Three MAAP analyses were performed to develop containment pressures, and these were convoluted with the containment fragility distribution to assess the probabilities of the various containment failure modes. As was mentioned for the very early failure case, the small number and range of MAAP analyses employed here is considered to be weak.

Event 6, CET:PLBYP: "Pool Bypass At/Near RV Failure"

Suppression pool bypass can occur in a number of ways at River Bend, and could result in unattenuated fission products being transported into the containment atmosphere. This node sorts the PDS into those events where pool bypass occurs and those where pool bypass does not occur. The DET (subtree) for this node consists of the following three DET top events:

1. DET(PLBYP) - LOCATRAN "LOCA or Transient": This node is quantified based a PDLD node. The two possible branch outcomes are: (1) LOCA; or (2) Transient.

2. DET(PLBYP) - SBO "SBO or Non SBO": This node is quantified based on a PDLD node. The two possible branch outcomes are: (1) SBO; or (2) no SBO.

3. DET(PLBYP) - PLBYP "Pool Bypass Before Vessel Failure": This node is quantified based on engineering judgment. It defines the outcome of this DET, as either: (1) no pool bypass (P=0.8); or (2) pool bypass (P=0.2).

Event 7, CET:INJ-FAIL: "Early Containment Failure Fails All In-Vessel Injection"

This node evaluates the effects of containment failure on the continued availability of injection system piping, pumps, and valves after the time vessel failure occurs. The DET is similar to DET(INJ_FAL1), except now drywell failure is an additional potential cause of failure of injection piping. The two possible outcomes of this node are: (1) no injection failure; or (2) injection failed. The DET (subtree) for this node consists of the following four DET top events:

1. DET(INJ_FALL) - CF_MODE "Containment Failure Mode": This branch sorts the early containment failure modes (gross or penetration failure). It is quantified based on node CET:ERLY_CF. The possible branches for this sub-node are: (1) gross failure; or (2) penetration failure.

2. DET(INJ_FALL) - DW_FAIL "Drywell Failure Occurs": This branch is quantified by CET:EARLY_DW, and quantifies the possibility of realizing the following outcomes: (1) drywell failure, or (2) no drywell failure.

3. DET(INJ_FALL) - PIPE_FAIL "Injection Piping Not Disrupted": This branch in the DET determines if injection piping fails due to gross containment failure or penetration failure, and is similar to DET(INJ_FAL1):PIPE_FAIL. The possible branches for this sub-node are: (1) piping not disrupted (P=1.0, penetration failure; P=0.1, gross failure); or (2) piping disrupted and injection failed (P=0.0, penetration failure; P=0.9, gross failure). The values for split fractions assigned are based on engineering judgment.

4. DET(INJ_FALL) - INJ_FALL "Injection Power Supply/Motor/Pumps Not Failed": This branch defines the result from this DET. It is quantified based on engineering judgment. The possible branches for this sub-node are: (1) pumps and power supplies available (P=0.5); or (2) failed, not available (P=0.5). A sensitivity analysis is conducted in the IPE for these split fractions. It is not clear from the IPE how the split fractions for this node were derived.

Event 8, CET:ENTRAIN: "Fraction of Core Debris Transported to Drywell"

This DET considers the fraction of debris transported to the drywell at the time of vessel failure. This node impacts the potential for direct containment heating and for core-concrete interaction. The two possible outcomes of this node are: (1) high (~40% of core inventory transported to the drywell); or (2) low (~5% of core inventory transported to drywell). The DET (subtree) for this node consists of the following five DET top events:

1. DET(ENTRAIN) - RXPRESS "RPV Pressure At RPV failure": (Same as Node 4, Sub-tree 1).

2. DET(ENTRAIN) - LAT_INJ "Late In-Vessel Injection": (Same as Node 1, Sub-tree 2).

3. DET(ENTRAIN) - FR_MOBILE "Fraction of Core Debris Mobile at RV Breach": This node sorts accident progression into sequences having large and small amounts of core debris mobile at the time of RV breach, based on the outcome of LATE_INJ. This node is quantified using NUREG/CR-4551 results. The possible branches for this sub-node are: (1) high mobile fraction (~40% of core debris mass); or (2) low mobile fraction (~10% of core debris mass).

4. DET(ENTRAIN) - HPME "High Pressure Melt Ejection Occurs": This node sorts the CET sequence categories associated with occurrence and non-occurrence of high pressure melt ejection. HPME requires a HP vessel at the time of vessel breach. This node is quantified loosely based on NUREG/CR-4551 results. The possible branches for this sub-node are: (1) HPME occurs (P=0.8); or (2) no, HPME does not occur (P=0.2).

5. DET(ENTRAIN) - ENTRAIN "Fraction of Core Debris Transported to Drywell": This node defines the result from this DET. It is quantified based on engineering judgment. The split fractions depend on the fraction of core debris mobile at RV breach and on the outcome of HPME (see p. 627 of IPE). The possible branches for this sub-node are: (1) high, (~40% of core debris mass transported); or (2) low (~5% of core debris mass transported).

Event 9, CET:EXV-COOL: "Debris Cooled Ex-Vessel"

This CET node investigates whether or not the molten core debris can be cooled ex-vessel, thus averting significant concrete ablation. The two possible outcomes of this node are: (1) debris can be cooled ex-vessel (concrete ablation averted); or (2) debris cannot be cooled ex-vessel (concrete ablation occurs). The DET (subtree) for this node consists of the following four DET top events:

1. DET(EXV_COOL) - RXPRESS "RPV Pressure At Failure": (Same as Node 8, Sub-tree 1).

2. DET(EXV_COOL) - LAT_INJ "Late In-Vessel Injection": (Same as Node 8, Sub-tree 2).

3. DET(EXV_COOL) - DEB_PED "Fraction of Core Debris Entrained from Pedestal": This node quantifies the amount of core debris entrained from the pedestal. This node is quantified based on CET:ENTRAIN. The possible branches for this sub-node are:

(1) high, (~40% of core debris mass entrained); or (2) low (~5% of core debris mass entrained).

4. DET(EXV_COOL) - EXV_COOL "Debris Cooled Ex-Vessel": This node defines the result from this DET. It is quantified based on NUREG/CR-4551 data from the analysis of Grand Gulf. The split fractions depend on outcomes of RXPRESS, LAT_INJ and DEB_PED (see p. 628 of IPE). The possible branches for this sub-node are: (1) debris cooled ex-vessel; or (2) debris not cooled ex-vessel.

The CET analysis presumes that no water will be present in the reactor pedestal cavity at the time of debris expulsion, because of the water-tight pedestal door. Hence, the potential for a steam explosion within the cavity does not appear to be fully addressed. The assumption in the IPE that the water-tight door has similar capacity as the pedestal itself does not seem reasonable, and no significant justification (e.g., based on an analysis of its capacity) is provided for this assumption. The treatment in the RBS IPE of impact of steam explosion, therefore, is considered weak.

Event 10, CET:LATE-CF#: "Mode of Late Containment Failure"

This node is slightly different than the previous CET nodes related to very early and early containment failures. It determines the conditional probability of each potential late containment failure mode that may occur as a result of late hydrogen combustion, containment over-temperature, long-term containment over-pressure, or basemat melt-through. Hydrogen detonation is no assumed to be longer a concern at this stage of the analysis. The four possible outcomes of this node are: (1) gross failure; (2) penetration failure; (3) no failure; or (4) basemat melt-through. The DET (subtree) for this node consists of the following nine DET top events:

1. DET(LATE_CF) - PREV_BURN "Did a Hydrogen Burn Occur That Did Not Fail Containment": This node sorts CET sequences into events where previous hydrogen burns have, or have not, occurred. The node is quantified based on CET:V_ERLY_CF and CET_EARLY_CF. The two possible branch outcomes are: (1) previous burn occurred; or (2) no previous burn occurred.

2. DET(LATE_CF) - CHR "Is Late Containment Heat Removal Available": This node determines if CHR is available for PDSs where electric power has been recovered. CHR in this node is through Suppression Pool Cooling (SPC) only. The availability of late containment heat removal is quantified based on PDS:CHR. The possible branches for this sub-node are: (1) late SPC is available; or (2) no, late SPC is not available.

3. DET(LATE_CF) - LATE_INJ "Is Late Injection Available": This node determines if there is availability of water injection for PDSs where electric power has been recovered. The purpose is to separate PDSs where containment injection is recovered

after vessel failure but prior to containment failure. This sub-node is quantified based on PDS:LATINJ. The possible branches for this sub-node are: (1) late injection is available; or (2) late injection is not available.

4. DET(LATE_CF#) - VENT "Is Long-Term Containment Venting Available": This node determines if the containment venting system is available to prevent long term pressurization of the containment. Containment venting at River Bend is through a 3-inch vent line. This sub-node is quantified based on PDS:VENT. The possible branches for this node are: (1) venting is available; or (2) venting is not available.

5. DET(LATE_CF#) - DBCL_CCI "Does Core-Concrete Interaction Occur": This node sorts the CET sequences based on debris coolability; events where debris is not coolable will involve significant core-concrete interactions, with resulting non-condensible gas generation that may fail the containment. This sub-node is quantified based on split fractions from NUREG/CR-4551, and depends on the previous status of CHR and LATE_INJ. The possible branches for this sub-node are: (1) no CCI occurs; or (2) CCI occurs.

6. DET(LATE_CF#) - LARGE_BRN "Detonation of Large Burn": This node sorts the CET sequences into the most likely type of hydrogen burn, based on hydrogen and steam concentrations in the containment. This sub-node is quantified based on split fractions from NUREG/CR-4551, and depends on previous status of CCI. The possible branches for this sub-node are: (1) detonation; or (2) large burn.

7. DET(LATE_CF#) - HBRN_RES "Mode of Containment Failure Due to Hydrogen Burn": This node sorts the CET sequences into the most likely containment failure modes based on expected peak pressures developed from hydrogen burn or detonation. Two MAAP analyses were performed to estimate containment peak pressures, in order to quantify this sub-node (using the containment fragility analysis). The possible branches for this sub-node are: (1) gross failure; (2) penetration failure; or (3) no containment failure.

8. DET(LATE_CF#) - CONT_BMM "Containment Failure Due to Basemat Melt-Through": This node sorts the CET end-states with no heat removal or water injection available into events where either basemat melt-through occurs or does not occur. The quantification of this sub-node is based on engineering judgment. The possible branches for this sub-node are: (1) no basemat melt-through; or (2) basemat melt-through.

   The IPE team used a split fraction of (1.0, 0.0) for this node. Although, their rationale for this assessment is clear, the result is questionable. In reality, there will be some probability for basemat melt-through. The justification for the development of this (and other) split fractions that involve engineering judgment are, in some cases, weak.

9. DET(LATE_CF#) - LATE_CF# "Mode of Late Containment Failure": This final branch in the DET sorts sequences into the potential containment failure modes, based on results of HBRN_RES and CONT_BMM. The possible branches for this sub-node are: (1) gross failure; (2) penetration failure; (3) no failure; or (4) basemat melt-through.

   As was noted previously, the number and description of MAAP runs for this case are somewhat inadequate.

In general, the DET/CET analysis for River Bend is judged to be a logical approach for modeling severe accident progression at the plant. As already stated, the analysis is somewhat similar to the NUREG-1150 APET analysis for Grand Gulf, although some additional modeling details are provided in the River Bend analysis. The CET and DETs for the River Bend analysis are provided in Figure 4.5-1 and Figures 4.6-1 to 4.6-11 of the IPE.

2.2.3 Containment Failure Modes and Timing

A plant-specific evaluation of the structural capacity of the RBS Mark III containment was conducted for the IPE, and is summarized in Section 4.4 of the IPE submittal. This analysis deals exclusively with integrity of the primary containment (steel vessel and drywell), including major structures, airlocks, hatches, and penetrations. The specific containment failure modes evaluated in the IPE include:

- Gross structural failure of the containment vessel, including the reactor pedestal
- Failure of the steel containment vessel base anchorage
- Structural failure of the drywell
- Failure of major penetrations

All load conditions considered in the analyses were for quasi-static pressurization under various temperature conditions; dynamic effects were neglected. The containment structural failure assessment is, therefore, not relevant to dynamic loads resulting from energetic phenomena (e.g., fuel coolant interaction).

Structural integrity of the RBS primary containment (steel vessel, drywell, and pedestal) under overpressure-related severe accident load conditions was performed by EQE Engineering, Inc. Fragility assessments were performed, although only values for median capacities were presented in the IPE. The locations examined in this failure analysis included the following:

- Steel Containment Vessel;
- Drywell;
- Drywell Head;

- Drywell Combination Equipment Hatch and Personnel Door;
- Reactor Pedestal;
- Reactor Building Basemat (melt-through);
- Steel Containment Vessel Anchorage;
- Major Containment Penetrations;
- Containment Vessel Personnel Locks;
- Containment Vessel Equipment Hatch; and
- Containment Dome Vent.

Capacities were assessed for the following temperatures: 70°F, 300°F, and 800°F. Results for median capacities at 300°F are provided in Table 4.4-1 of the submittal. This table does not list capacities for all items discussed in the text. In addition, there are discrepancies between the values presented in Table 4.4-1 and those presented in the text (e.g., containment dome vent opening capacity), as well as the value for containment failure pressure presented in Table 4.4-1 and that in Table 4.1-2. Table 4.4-1 should also list values for variability parameters (β) that define fragility curves, and it would be useful to present plots of these fragility curves in the IPE for the various temperature conditions. The containment fragility analysis does not include assessment of the capability of the weir wall.

For the analysis of inflatable seal failures, the IPE states (e.g., see Section 4.4.3.2) that at higher temperatures (300°F $< T <$ 400°F) the seal inflation pressure gets higher so long as the seal integrity is maintained, such that the seal leakage pressure will be higher. In this regard, the IPE needs to provide additional explanation and justification as to why the seal failure/leakage pressure increases with temperature.

In general, the documentation in Section 4.4 of the IPE is weak. Results are inconsistent in places and the discussion is sometimes unclear. As an example, there is no discussion of uncertainty in the containment fragility analysis. Thus, it is unclear how the containment failure probability is estimated.

2.2.4 Containment Isolation Failure

In the River Bend IPE submittal, containment isolation failure was analyzed using the front-end fault tree analysis. The calculated conditional probability of isolation failure is 0.028. No other significant details are provided in the documentation. In response to an NRC question, the licensee stated that a systems analysis was performed for each isolation valve. 22 penetrations were identified as potential locations of isolation failure. The overall isolation failure probability was calculated to be 0.0046 per event. The licensee stated that " this value was conservatively increased to 0.0326 based on engineering judgement to account for uncertainties in Level 2 analysis". It is not known to this review why the uncertainties in the back-end analysis should affect the calculation of containment isolation failure.

2.2.5 System/Human Response

There is virtually no mention of Human Reliability Analysis (HRA) in the back-end analysis. Only one HEP is mentioned as being assessed in the entire Level-2 analysis. A human error analysis was conducted to evaluate the probability that the hydrogen igniters are not turned on, and then off, correctly. The HEP that the igniters are in the "on" condition when power is recovered was evaluated to be 0.07. This is the only occurrence of reference to human error analysis in the back-end portion of the IPE. Yet, this HEP result was apparently not even used in quantifying the DET/CET model (see pg. 607 of IPE). The licensee stated (in response to the NRC questions [10] that "given the level of resolution and uncertainty inherent in the Level 2 analysis, it is unreasonable to perform a detailed human reliability analysis".

2.2.6 Radionuclide Release Categories and Characterization

The end-states of the CET represent the entry points to the Source Term Logic Diagram (STLD). The STLD serves to group the end-states of the CET sequences into release categories with similarities in accident progression and in source term characteristics. The methodology of the STLD is similar to PDS binning. The important characteristics that define the release categories (i.e., nodes in the STLD) in the RBS IPE are the following:

- Containment Bypass: The first node in the STLD determines whether the radionuclide releases are due to bypass sequences. Although the Level-I PRA does not indicate any bypass sequence that survived the cutoff limit.

- Debris Coolability In-Vessel: The next node in the STLD determines whether debris is cooled in-vessel. The accident is effectively terminated with limited releases if debris is cooled in-vessel.

- Containment Isolation Status: This is considered to be an important attribute because fission products released during accident progression can find a direct path to the environment. The choice of the isolation path selected was the steam line isolation valve, since it was the largest diameter isolation failure pathway.

- Suppression Pool Bypass Failure: This characteristic differentiates between drywell failure and direct suppression pool bypass.

- Suppression Pool Temperature: This characteristic differentiates between saturated and subcooled pool, since the pool temperature has a direct impact on the fission product scrubbing.

- Ex-Vessel Cooling: Ex-vessel cooling has a major impact on the vaporization release, and chosen as a source term characteristic. A depth of a debris greater than 0.15 m is considered not coolable, and a debris bed of depth less than 0.05 m with overlying water layer is considered to be always coolable.

- Time of Containment Failure: Three different choices of containment failure are chosen, namely, very early, early, and late.

- Mode of Containment Failure: Four modes of containment failure are chosen, namely, penetration failure, gross failure, no containment failure, and venting.

The definition of radionuclide release categories in the RBS IPE is sufficiently detailed, and provides sufficient information on the releases. Although there are 768 possible outcomes, only 12 Source Term Categories (STCs) were found to be statistically significant. Fission product retention by auxiliary building is not considered in the analyses.

2.3 Quantitative Assessment of Accident Progression and Containment Behavior

2.3.1 Severe Accident Progression

MAAP-BWR 3.0B, Revision 7.02 was the principal tool used to analyze postulated severe accidents at River Bend. 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 GSU internal document.

A small number of simulations were performed using the MAAP code for various accident sequences (station blackout, transients). However, the submittal does not clearly list all the simulations performed (other than those few used to generate estimates of containment peak pressures), results of the simulations, and details of the sensitivity analyses. The IPE does state that the MAAP parameters were taken from Reference [6].

Other than River-Bend-specific MAAP analyses, quantification of DETs in the Level-2 IPE relies heavily on use of split fractions and other results developed in the NUREG-1150 analysis for Grand Gulf (NUREG/CR-4551). In some cases, it is questionable that the split fractions (and pedestal capacity) for Grand Gulf are applicable to River Bend. Aside from such cases, the usage of these results in the RBS IPE is thought to be generally appropriate. Other instances of quantification in the CET analysis are based solely on engineering judgment of the analysis team. In several instances, no justification is given for the split fractions selected on this basis. Many (but not all) of the split fractions involving engineering judgment are, however, further evaluated by means of sensitivity studies.

With exception of the cases identified, therefore, the quantification of accident progression in the RBS CET/DET analysis appears to be reasonable, logical and acceptable.

In addition to the base-case CET/DET quantification, the River Bend performs a number of sensitivity analyses, primarily associate with variations in assumed split fractions. Although several (23) cases were initially identified, some of these were subsequently incorporated into other sensitivities addressed by the IPE team. The impact of the sensitivity variations was measured by comparing results for frequencies of containment failure modes obtained for the base case against results obtained from the sensitivity variation. Sensitivity variations are

discussed in Section 4.6.11.4 of the submittal. Below are highlighted the 11 cases that alter the containment failure frequencies by more that (plus or minus) 10%. In addition to these cases, two sensitivities determined to have little effect on containment failure mode frequencies were evaluated for further (potentially significant) impacts on source terms.

1. In-Vessel Debris Cooling. A 4% increase in penetration failure frequency, less than 1% increase in containment survivability, and a 13% reduction in gross containment failure frequency was obtained for the case where debris is never cooled in-vessel. The base case assigned split fractions to debris cooling based on size of vessel breach.

2. Amount of Zircaloy Reacted In-Vessel. A 13% increase in gross containment failure for 30% MWR; 28% decrease in gross containment failure for 20% MWR; and a 83% decrease in gross containment failure for 10% MWR. For 10% MWR, there was also a 10% increase in penetration failures and a 12% increase in containment survivability. The base case assigned split fractions of (0.05, 0.20, 0.75) for (10%, 20%, 30%) MWR.

3. Random Hydrogen Ignition and Burn Size on Early Containment Performance. Results for this sensitivity case are shown in Table 4.6-2 of the IPE for several cases of variation in split fractions for (Ignition/No Ignition) and (Burn/No Burn). Impacts on gross containment failure frequency varied from -20% to +10%; for penetration failures, the frequency changes varied from -17% to +6%; and changes in containment survivability fraction varied from -5% to +11%.

4. Very Early Hydrogen Detonations or Large Burns on Very Early Containment Performance. 120% increase in gross containment failure frequency, 16% decrease in containment survivability, and 17% decrease in penetration failure frequency for the case when detonations occur 100% of the time. 83% decrease in gross containment failure frequency, 11% increase in containment survivability, and 12% increase in penetration failure frequency for the case when burns occur 100% of the time. The split fraction for (Detonation/Burn) in the base case analysis depended on the result of DET(V_ERLY_CF):H2_INVESS.

5. Mean Pressure Demand on Very Early Containment Performance. The base-case mean pressure demand (determined from MAAP runs) was 25 psig. The variations of mean pressure demand are 30 psi and 20 psi. For the 30-psi case: gross containment failure frequency increased less than 1%; penetration failure frequency increased 23%; and containment survivability decreased 10%. For the 20-psi case: gross containment failure frequency decreased less than 1%; penetration failure frequency decreased 24%; and containment survivability increased 10%.

6. Hydrogen Detonations or Large Burns on Containment Performance. This sensitivity variation is similar to case (4) above. 18% increase in gross containment failure frequency, less than 1% increase in containment survivability, and 7% decrease in

penetration failure frequency for the case when detonations occur 100% of the time. 15% decrease in gross containment failure frequency, less than 1% increase in containment survivability, and 6% increase in penetration failure frequency for the case when burns occur 100% of the time. The split fraction for (Detonation/Burn) in the base case analysis depended on the result of DET(EARLY_CF):H2_INVESS.

7. Time of Injection Failure on Containment Performance. Time of injection failure impacts the steam concentration developed in the containment, which (in turn) impacts hydrogen detonation and burn potentials. The base case analysis used split fractions for various steam concentrations. In the sensitivity analyses, three cases of steam concentration were considered: 0-35%, 35-55%, and over 55%. For the 0-35% case: gross containment failure frequency increased 115%; penetration failure frequency decreased 19%; and containment survivability decreased 14%. For the 35-55% case: gross containment failure frequency decreased 84%; penetration failure frequency increased 14%; and containment survivability increased 10%. For the >55% case: gross containment failure frequency increased 109%; penetration failure frequency increased 41%; and containment survivability decreased 39%.

8. The IPE submittal states that this sensitivity case, denoted V1/V2, was included in the sensitivity case (10) below, denoted Y1/Y2 in the IPE.

9. Hardened Containment Vent (10-inch) on Containment Performance. This sensitivity case investigates the effect of increasing the containment vent path from a 3-inch line (base case) to a 10-inch line. This is a CPI issue. The impact of this changes is estimated as follows: 99.8% decrease in gross containment failure frequency, less than 1% increase in containment survivability, less than 1% increase in penetration failure frequency; and addition of a venting path for significant releases having a frequency of 1.84×10^{-6} (which is equal to the decrease in frequency of gross containment failures). Hence, the effect of increased vent path is to replace an uncontrolled large release with a controlled large release.

10. Alternative Power to Hydrogen Igniters. This is also a CPI issue. This sensitivity case investigates the effect of providing an alternate power supply to hydrogen igniters. Two cases are analyzed: (1) where alternate power is supplied; and (2) where no hydrogen igniters are present at all. The impact of case (1) is estimated as follows: 14% decrease in gross containment failure frequency, 7% increase in containment survivability, and 9% decrease in penetration failure frequency. The impact of case (2) is estimated as follows: 78% increase in gross containment failure frequency, 37% decrease in containment survivability, and 51% increase in penetration failure frequency.

    Partially based on these results, the licensee concludes that a plant change in response to this CPI is not warranted.

11. Improved SBO Injection Capability. This is a Mark-I CPI issue that has initial (and significant) impacts on the Level-1 analysis. The impacts are here continued, and propagated through the Level-2 analysis. The impact of improved SBO capability is: 89% decrease in gross containment failure frequency, 79% increase in containment survivability, and 91% decrease in penetration failure frequency.

From this analysis, and from the fact that the Level-1 risk is also significantly reduced, the licensee concludes that improving SBO injection capability is more effective than addressing the previous two CPI issues (which essentially impact only the Level-2 analysis). This conclusion is judged to be a valid and significant insight derived from the Level-2 analysis.

The two sensitivity cases (mentioned previously) that do not impact containment failure itself, but do have an impact or radiological releases, are not discussed further in the RBS IPE.

2.3.2 Dominant Contributors to Containment Failure

Table 2 shows a comparison of the conditional probabilities of the containment failure modes provided in the River Bend IPE submittal, together with the results of the IPE submittal for the Perry plants, and the NUREG-1150 study for Grand Gulf [7]. All comparisons are made for internal initiating events only.

From a review of Table 2, it can be seen that the conditional probabilities of early (0.27) and late containment failure (0.114) for the RBS IPE are both smaller than the results for the Grand GULF/NUREG-1150 analyses. However, the dominant contributor to core damage in both plants are the same (SBO sequence).

Table 2 Containment Failure as a Percentage of Internal Events CDF: Comparison with Other PRA Studies

Containment Failure Mode	River Bend	Perry	Grand Gulf
Early Failure	27	24.8	44.8
Late Failure	11.4	7.4	28.4
Venting	0	29.3	3.8
Intact	61.6	39.1	23.0
Core Damage Frequency, yr^{-1}	$1.55x10^{-5}$	$1.3x10^{-5}$	$4.1x10^{-6}$

It appears that there are two reasons for the differences. First, there are plant differences. The River Bend station has a containment (1.2×10^6 ft^3) and drywell (251,000 ft^3), whose volumes are comparable to the Grand Gulf plant (1.4×10^6 ft^3 and 270,000 ft^3, respectively), however, the reactor power in River Bend (2894 Mwt) is much less than Grand Gulf (3833 Mwt). Accordingly, there is 25% less Zircaloy and UO_2 in the River Bend plant. This has an important impact on accident progression since hydrogen generation and combustion dominates containment failure in BWRs with Mark III containment. The hydrogen concentration in the River Bend containment can be expected to be 20-25% less than Grand Gulf values. So, it is anticipated that the conditional probability of containment failure in RBS should be less than Grand Gulf.

The other important difference pertains to the CET analyses in River Bend plant. In-Vessel recovery of degraded core (due to AC power recovery and subsequent injection) is treated differently in the two studies. In the River Bend IPE submittal, the licensee calculates a rather high probability of AC power recovery before vessel breach (the power recovery probabilities are listed in Table 4.7-5 of the submittal, for most PDSs it seems to be around 0.85). Although the subsequent treatment of core damage arrest by reflood is identical between the IPE and NUREG-1150, the results are dominated by AC power recovery models. The results show that, the conditional probability (given core damage) of core damage arrest in-vessel is 58% in the IPE submittal, as compared to about 18% in NUREG-1150. Of course, core damage arrest does not necessarily mean no containment failure. Hydrogen generated before in-vessel recovery can still fail the containment, and in the submittal, the conditional probability of containment failure with core damage arrest (in-vessel) is 22%. However, the rest of the sequences in the submittal (corresponding to 36% of the CDF) entail core damage arrest in-vessel, no vessel breach, and no early or late containment failure. These sequences represent the difference in the results between the two studies.

Another minor difference pertains to the calculated containment capacity of the two plants. NUREG-1150 analysts estimated a mean failure pressure of 41 psig for Grand Gulf, whereas the IPE calculates a (minimum) failure pressure of 56 psig for the free-standing steel RBS containment. This may also account for some of the difference in the results.

The IPE estimates that among all core-damage sequences, 61.6% result in no containment failure, 26.8% result in penetration failures (including 2.8% due to failure to isolate), and 11.6% result in gross (anchor/dome) failure. The highest radioactive release fractions were found to be associated with events where there is a SBO and a loss of containment isolation. In addition, only SBO events were found to lead to gross (vessel anchor or dome) containment failure. A total large release frequency of 1.84×10^{-6} per year is estimated for RBS due to internal initiators (based on the point-estimate analysis).

2.3.3 Characterization of Containment Performance

Containment performance is characterized in the River Bend IPE by means of estimates of frequencies of failure for the significant potential containment failure modes considered (gross

failure, penetration failure, basemat melt-through (zero estimated frequency of occurrence), and no failure). These containment failure frequencies are based on consideration of APB frequencies obtained from the CET quantification analysis. Containment performance is also characterized in the IPE by large release frequency, where large release is defined simply as one involving gross containment failure.

2.3.4 Impact on Equipment Behavior

The IPE considered the impact of injection piping disruption (under containment failure) on response of equipment subjected to resulting harsh conditions. Failure was assumed for certain power supplies/motors/pumps under such conditions. No other impact of harsh conditions on equipment response was apparently considered. It is uncertain whether or not the IPE considers all equipment that may be effected by injection piping disruption.

2.4 Reducing the Probability of Core Damage and Fission Product Releases

2.4.1 Definition of Vulnerability

The licensee defines "vulnerability" at the plant level, based on comparison of plant CDF with the NRC subsidiary safety goal of 1×10^{-4} mean annual rate of core-damage per reactor. On this basis, the licensee concludes that there are no vulnerabilities at RBS, RBS is not a risk outlier, and RBS does not constitute an undue risk to public health and safety.

The IPE submittal states that RBS does not meet a large release safety goal of 1×10^{-6}. In response to NRC questions, the licensee stated that no vulnerability exists based on "the small changes in containment failure for the sensitivity cases performed for the Level 2 IPE". The licensee stated that the large containment frequency is driven by the fact that the core damage frequency is primarily due to station blackout. Plant modifications made after the completion of the IPE, have reduced the CDF from 1.55×10^{-5} per reactor year to 3.55×10^{-6} per reactor year. Hence, the licensee apparently concludes that no vulnerabilities exist in the RBS plant.

2.4.2 Plant Modifications

As discussed previously, in regard to plant improvements that are relevant to the Level-2 analysis, the RBS IPE evaluates the risk impacts of two Mark III CPI issues: (1) providing for increased containment vent capability, and (2) providing an alternate power supply for hydrogen igniters. In addition to these considerations, the IPE evaluates the Level-2 impact of improving SBO injection capability and enhanced depressurization. Improving SBO injection capability requires the following plant changes:

1. Providing a portable diesel generator and means of connection to DC power bus.

2. Remove check-valve internals in FPW/SSW cross-tie.

3. Revise SBO procedure to provide a different FPW injection path, using valves outside containment.

These changes were determined to have the following beneficial impact on Level-2 risk: gross containment failure-risk contribution is reduced 35% (from 11.6% to 7.5% of all core damages); containment penetration failure-risk contribution is reduced 43% (from 26.8% to 15.2%); and relative fraction of containment survivability increases 25% (from 61.6% to 77.3%). In addition, core-damage frequency decreases significantly from 1.55×10^{-5} to 3.4×10^{-6}. These modifications have been implemented after the IPE was submitted.

2.5 Responses to CPI Program Recommendations

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 III containments, the following improvements were identified:

1. Installation of a hardened vent.
2. Alternative water supply for drywell spray/vessel injection,
3. Enhanced reactor pressure vessel depressurization system reliability,
4. Implementation of Revision 4 of the BWR Owners Group EPGs, and
5. Improved hydrogen igniter power supply.

Alternative Water Supply for Drywell/Spray Injection: Alternative water for RPV injection can be provided in the RBS by the diesel driven Fire Protection Water (FPW) pump. The pump can be aligned via the SSW crosstie to the RHR for injection into the vessel. Since this is a low pressure pump, the reactor must be depressurized to allow injection. When DC power is lost as in a long-term SBO, control of SRVs is lost, and FPW is lost as a source injection to the vessel. Thus an alternate source of DC power is needed. A portable DC generator can provide extended DC power during a SBO.

Three plant changes were made after the IPE submittal was completed to realize this alternate water supply:

- A portable DC generator was added to provide DC loads for SBO.
- Internals from three check valves in the FPW/SSW cross-tie were removed, and manual gate valves were added.
- A SBO procedure was modified to provide a different FPW injection path for SBO which uses valves outside the containment.

The impact of these modifications on the CDF was found to reduce the CDF to 3.55×10^{-6} per reactor year. The impact of these modifications on the results of containment analyses and radionuclide releases are calculated, by propagating through the Level-2 analysis. The impact of improved SBO capability is calculated to be the following: 35% reduction in gross containment failure frequency, 43% reduction in the frequency of penetration failure, the conditional probability of containment remaining intact increases from 0.616 to 0.773.

These plant modifications were made in 1993 after the IPE documentation was submitted to the NRC. Alternate injection to the drywell was not considered since the RBS plant does not have drywell sprays.

Enhanced reactor pressure vessel depressurization system reliability: The River Bend depressurization system consists of 16 safety relief valves. A Level-1 sensitivity analysis was performed for adding a station blackout diesel generator for enhanced reliability of the DC power to SRVs. Loss of instrument air to the SRVs during a station blackout was assessed based on the SRV accumulator size and the addition of two temporary diesel driven compressors. One of the two temporary compressors is to be installed in 1995. Thus, the licensee has addressed the need to enhance the reactor pressure vessel depressurization reliability.

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 RBS EOPs in 1987.

Hardened Vent: RBS currently has a 3-inch (hydrogen purge) vent line. This vent path was determined to be insufficient in averting containment over-pressure failure for events with loss of all containment heat removal. The vent size require to avert such failure was determined to be 10 inches. Hence, the risk impact of installing a 10-inch hardened vent line was evaluated in the IPE.

The level-1 impact of this potential plant change was studied by identifying those existing core-damage sequences that would be recovered as a result of the increased vent capability. Four such sequences were identified that had survived the initial truncation process. Cut sets for these four sequences were deleted from the overall core-damage cut sets listing, and a new estimate of CDF was made, to reflect the improved vent capability. The CDF was found to decrease only slightly, from 1.55×10^{-5} to 1.53×10^{-5}.

From a back-end perspective, the improved vent capability was judged in the IPE to bring about a trade-off in its impact. The beneficial impact of this potential plant change, as evaluated in the submittal, is to essentially eliminate gross containment failure. On the other hand, the increased vent capability was found to lead to large radiological releases (albeit monitored releases). The reduction in the magnitude of radioactive releases is associated with the elimination of gross containment failure is, therefore, suggested in the IPE submittal, to be accompanied by a roughly equal increase in radioactive release. The IPE implies that increasing hardened vent capability is not warranted. The licensee's analysis of the increased hardened vent capability provides some meaningful insights on the CDF. However, the licensee evaluation of

the impact on containment failure and radionuclide releases, is purely hypothetical. The IPE itself acknowledges that radioactive releases through the increased hardened vent would be monitored and could be isolated at some point during the severe accident. Operator action would be involved in isolating the vent. Installation of increased hardened vent capability would probably not eliminate the risk of gross containment failure. Containment failure pressure is not known with sufficiently high certainty to be able to say that a 10-inch vent would completely preclude the possibility of containment failure; but, even if a 10-inch vent path was sure to avert containment failure, the operator may not be able to (or may not) open the vent path. Thus it would have been worthwhile on the part of the licensee to assess the impact of the increased hardened vent capability.

Improved AC Power Supply to the Hydrogen Ignitors: The RBS IPE evaluated the potential modification to the electric supply to the hydrogen igniters that would help ensure function of the igniters under SBO conditions. Such modification could possibly reduce the possibility of high containment loads that may otherwise be realized from hydrogen deflagrations and detonations. The impact of this change was found to reduce the percentage conditional probability of penetration failure by 9.7% (from 26.8% to 24.2%) and to reduce the conditional probability of containment structural failure by 12% (from 11.6% to 10.2%). On this basis, the IPE concludes that a hardware upgrade to provide uninterrupted power supply to the hydrogen igniters is not warranted. Related to the issue of power supply to hydrogen igniters, however, the IPE does suggest the need for change to abnormal operating procedure AOP-0050, which applies during SBO events. The revision to this AOP instructs the operators to turn off the igniters if AC power is unavailable.

Although the IPE provides significant insights with respect to impacts of providing alternate power for the hydrogen igniters, we believe this issue is worthy of further analysis. As the IPE itself mentions (Section 6.2.3) and as pointed out by the licensee's reviewers (Sections 5.3 and 5.4), additional evaluation of hydrogen igniter use under SBO scenarios is required.

In response to the NRC questions, the licensee concluded that increasing the hardened vent capacity and provision of alternate power source for the hydrogen ignitors, was not cost-effective.

3. CONTRACTOR OBSERVATIONS AND CONCLUSIONS

In summary, the River Bend back-end IPE analysis represents a major modeling and computational effort. The analysis is, in general, logically prepared and well executed (with certain exceptions, as noted). The study produces reasonably meaningful insights as to the nature and frequency of severe-accident progressions and radiological release risks for the plant.

The back-end portion of the RBS IPE submittal provides a substantial amount of information in regard to the subject areas identified in Generic Letter 88-20 and NUREG-1335. 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 III containments. The treatment of phenomenologic issues in the CET is quite detailed, and the IPE makes use of results from NUREG-1150 analyses. As an overall observation, the RBS back-end analysis follows closely the methodology and inputs used in NUREG-1150 for Grand Gulf. The major differences between the River Bend and Grand Gulf plants (i.e., those noted in Section 2.2) have been reasonably well accounted for in the RBS Level-2 model. The minor differences (e.g., reactor pedestal pressure capacity) have, in some instances, not been meaningfully modeled. Sensitivity studies have been undertaken by the licensee to assess changes in containment failure probabilities associated with a number of uncertain parameters used as inputs in the CET split fraction quantification process.

The most significant finding of the licensee is the possible reduction in CDF (and in radiouclide release frequency) due to provision of alternate injection and back-up power source in SBO sequences. The impact of these modifications on the CDF was found to reduce the CDF to 3.55 x 10^{-6} per reactor year. The impact of these modifications on the results of containment analyses and radionuclide releases are calculated, by propagating through the Level-2 analysis. The impact of improved SBO capability is calculated to be the following: 35% reduction in gross containment failure frequency, 43% reduction in the frequency of penetration failure, the conditional probability of containment remaining intact increases from 0.616 to 0.773. These modifications (i.e., installation of portable DC generator, removal of internals from three check valves in the FPW/SSW cross-tie, and addition of manual gate valves, and modification of a SBO procedure to provide a different FPW injection path for SBO which uses valves outside the containment) were made after the IPE documentation was submitted to the NRC.

A number of minor weaknesses were identified in the submittal, and they include the following:

- The number and variety of deterministic severe accident analyses (i.e., MAAP simulations) are limited.

- The analysis models basemat melt-through, but completely eliminates its possibility in quantification.

- The treatment of ex-vessel steam explosion in the reactor cavity is weak. The water-tight door cannot just be assumed to have the same capacity as the pedestal (whose capacity is not calculated); there is some probability that the door will not keep water out of the pedestal cavity, and thereby the potential for ex-vessel steam explosions cannot be ruled out.

- There is minimal or no treatment of operator actions in back-end analyses.

- The meaning of the IPE's definition of large release is somewhat arbitrary. Meaningful summaries (i.e., not just the 59 STC frequencies) pertaining to the timing and magnitude of radioactive releases are not reported.

- When split fractions are developed for the DET/CET quantification based on engineering judgment, the rational of the quantification (or the choice) of the split fractions are often not presented.

The important points of the technical evaluation of the River Bend IPE back-end analysis are summarized as follows:

- The back-end portion of this IPE submittal, for the most part, is relatively well performed, and provides a substantial amount of information in regard to the subject areas identified in Generic Letter 88-20 and NUREG-1335.

- The submittal includes most phenomena of interest to severe accident phenomenology for BWRs with Mark III containments; the level of CET modeling detail is apparently comparable to that for the NUREG-1150 analysis of Grand Gulf.

- The treatment of phenomenologic issues in the CET is mostly reasonable, and the IPE makes good use of results from NUREG-1150 analyses, and from NRC-sponsored research.

- The licensee evaluated meaningful plant improvements, and implemented one group of plant modifications, which lead to calculated reductions in CDF, frequencies of containment failure and radionuclide release.

4. REFERENCES

1. "River Bend Station Individual Plant Examination," Gulf States Utilities Company, February 1, 1993.

2. USNRC, "Individual Plant Examination: Submittal Guidance," NUREG-1335, August, 1989.

3. USNRC, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR §50.54(f)," Generic Letter 88-20, November 23, 1988.

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

5. USNRC, "Evaluation of Severe Accident Risks: Grand Gulf, Unit 1," NUREG/CR-4551, Rev. 1, December 1990.

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

7. USNRC, "Policy Statement on Severe Reactor Accidents Regarding Future Design and Existing Plants," Federal Register - Vol. 50, p. 32138, August 8, 1985.

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. Responses to NRC Request for Additional Information on RBS IPE, Attachment to Letter from James J. Fisicaro to the U.S. Nuclear Regulatory Commission, Dated September 22, 1995.

APPENDIX A

IPE EVALUATION AND DATA SUMMARY SHEET

BWR Back-End Facts

Plant Name

River Bend Station

Containment Type

Mark III

Unique Containment Features

- Use of containment fan coolers, rather than containment sprays, for containment heat removal;
- No vacuum breakers or check valves between drywell and containment air volumes;
- No provision for suppression pool make-up via upper fuel-pool dump;
- Water-tight access door to reactor pedestal cavity;
- Containment and drywell hydrogen igniters are powered by separate electrical circuits;
- Annular space between containment vessel and shield building, filtered by standby gas treatment system, for increased decontamination of radioactive releases;
- Small (3-inch) containment vent path, insufficient to avert containment over-pressure failure; and
- A 30-inch containment dome vent which was permanently sealed (welded) after construction in such a manner as to create significant local stress risers

Unique Vessel Features

None found

Number of Plant Damage States

43

Containment Failure Pressure

Median failure pressures are 67 psig, 60 psig, and 45 psig for local dome-vent failure at temperatures, respectively, of 70°F, 300°F and 800°F. For gross membrane failure of the steel cylinder, median pressure capacities are 125 psig, 111 psig and 84 psig for the preceding temperatures. Capacities are somewhat lower (e.g., 107 psig at 300°F) for gross membrane failure of the containment dome. Capacities of hatches and airlocks are lower still.

Additional Radionuclide Transport and Retention Structures

Shield building (and volume between containment vessel and shield building), auxiliary building, and fuel building

Conditional Probability That the Containment Is Not Isolated

2.8%

Important Insights, Including Unique Safety Features

See Executive Summary and Section 3 of this review.

Implemented Plant Improvements

A Level-1 improvement was implemented related to installation of SSW return valve; three procedural changes related to the Level-1 analysis were also implemented.

Improved SBO injection capability requiring the following plant changes:

1. Providing a portable diesel generator and means of connection to DC power bus.

2. Remove check-valve internals in FPW/SSW cross-tie.

3. Revise SBO procedure to provide a different FPW injection path, using valves outside containment.

No improvements related to CPI issues of increased hardened vent capability and of alternate power to hydrogen igniters were implemented.

C-Matrix

(43×59) matrix for 43 plant damage states and 59 source-term categories is too large to be provided here.