19. PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT

19.0 Background

The purpose of the staff's review of the probabilistic risk assessment (PRA) and severe accident evaluation is to ensure that the applicant has adequately addressed the objectives established by the Commission. These objectives are drawn from, "Licenses, Certifications, and Approvals for Nuclear Power Plants," of the Title 10 of the *Code of Federal Regulations* (10 CFR Part 52); the Commission's Severe Reactor Accident Policy Statement regarding future designs and existing plants; the Commission's Safety Goals Policy Statement; and the Commission-approved positions concerning severe accident requirements for advanced reactors contained in SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," and other documents. The objectives reflect the Commission's interest in the use of PRA in regulatory activities as indicated in the policy statement "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities." Specifically, the Commission has stated the objectives in numerous U.S. Nuclear Regulatory Commission (NRC) statements and Commission guidance, including the following:

1. NRC Policy Statement, "Severe Reactor Accidents Regarding Future Designs and Existing Plants," Volume 50, page 32138, of the *Federal Register* (50 FR 32138), dated August 8, 1985,

2. NRC Policy Statement, "Safety Goals for the Operations of Nuclear Power Plants," 51 FR 28044, dated August 4, 1986,

3. NRC Policy Statement, "Nuclear Power Plant Standardization," 52 FR 34884, dated September 15, 1987,

4. NRC Policy Statement, "The Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," 60 FR 42622, dated August 16, 1995,

5. SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," dated January 12, 1990, and the related staff requirements memorandum (SRM), dated June 26, 1990,

6. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," dated April 2, 1993, and the related SRM, dated July 21, 1993,

7. SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," dated June 12, 1996, and the related SRM, dated January 15, 1997, and

8. SECY-97-044, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," dated February 18, 1997, and the related SRM, dated June 30, 1997.

The first four NRC policy statements give guidance regarding the appropriate way to address severe accidents and use PRA. The Commission SRMs relating to SECY-90-016, SECY-93-087, SECY-96-128, and SECY-97-044 provide Commission-approved guidance for implementing features in new designs to prevent severe accidents and to mitigate their effects, should they occur.

With regard to PRA and severe accident evaluations, 10 CFR Part 52 of the Commission's regulations, in effect at the time the economic simplified boiling-water reactor (ESBWR) application for design certification was submitted, required the following PRA and severe accident information in an application for design certification:

- 10 CFR 52.47(a)(ii)—information with respect to compliance with a number of the technically relevant positions of the Three Mile Island requirements in 10 CFR 50.34(f)

- 10 CFR 52.47(a)(iv)—proposed technical resolutions of those unresolved safety issues and medium- and high-priority generic safety issues identified in the version of NUREG-0933, "A Prioritization of Generic Safety Issues," current on the date up to 6 months before the docket date of the application and technically relevant to the design

- 10 CFR 52.47(a)(v)—a design-specific PRA

19.1 <u>Probabilistic Risk Assessment</u>

19.1.1 Introduction

The following are the three main areas of the NRC review of the PRA and severe accident evaluation:

1. design-specific PRA
2. severe accident evaluations
3. application of results and insights of the design-specific PRA

The purpose of the staff's review is to ensure that the applicant has adequately addressed the objectives established by the Commission. These objectives include the following:

- Use the PRA to perform the following:

  – Identify and address potential design features and plant operational vulnerabilities, where a small number of failures could lead to core damage, containment failure, or large releases (e.g., assumed individual or common-cause failures (CCFs) could drive plant risk to unacceptable levels with respect to the Commission's goals, as presented below).

  – Reduce or eliminate the significant risk contributors of existing operating plants that are applicable to the new design, by introducing appropriate features and requirements.

  – Select among alternative features, operational strategies, and design options.

- Identify risk-informed safety insights based on systematic evaluations of the risk associated with the design such that the applicant can identify and describe the following:

– the design's robustness, levels of defense-in-depth, and tolerance of severe accidents initiated by either internal or external events

– the risk significance of potential human errors associated with the design

- Determine how the risk associated with the design compares against the Commission's goals of less than $1x10^{-4}$/yr for core damage frequency (CDF) and less than $1x10^{-6}$/yr for large release frequency (LRF). In addition, compare the design against the Commission's approved use of a containment performance goal (CPG), which includes (1) a deterministic goal that containment integrity be maintained for approximately 24 hours following the onset of core damage for the more likely severe accident challenges and (2) a probabilistic goal that the conditional core damage probability (CCDP) be less than 0.1 for the composite of all core damage sequences assessed in the PRA.

- Assess the balance between features of the design that prevent or mitigate severe accidents.

- Determine whether the plant design represents a reduction in risk compared to existing operating plants.[1]

- Demonstrate compliance with 10 CFR 50.34(f)(1)(i), which requires that a plant-specific PRA be performed to seek improvements in the reliability of core and containment heat removal (CHR) systems that are significant and practical.

- Use the PRA in support of the process employed to determine whether regulatory treatment of non-safety systems (RTNSS) is necessary and, if appropriate, the systems, structures, and components (SSCs) included in RTNSS.

- Use the PRA in support of programs associated with plant operations (e.g., technical specifications, reliability assurance, human factors, and maintenance).

- Use the PRA to identify and support the development of specifications and performance objectives for the plant design, construction, inspection, and operation, such as inspections, tests, analyses, and acceptance criteria (ITAAC), reliability assurance program (RAP), technical specifications (TSs), and combined license (COL) action items and interface requirements.

19.1.2 Quality of Probabilistic Risk Assessment

19.1.2.1 Summary of Technical Information

19.1.2.1.1 Description of the Probabilistic Risk Assessment

The ESBWR PRA is a full-scope (Levels 1, 2, and 3) PRA that covers both internal and external events for at-power and shutdown operations.

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[1] The reference to existing operating plants applies to LWR plant technology contemporary with the issuance of the Commission's Severe Accident Policy Statement on August 8, 1985.

The methodology used in the ESBWR Level 1 PRA is a linked fault tree approach. Fault trees have been developed and evaluated for the major ESBWR front-line and support systems to determine the probability that emergency core cooling and decay heat removal (DHR) systems perform their intended function when demanded. Transient and loss-of-coolant accident (LOCA) initiating events have been consolidated into major accident event sequences that are described by the accident event trees. These event trees are used to calculate the frequency of core damage sequences by directly linking the fault trees and solving for the minimal cutsets. Outcomes of the event trees are transferred to containment event trees (CETs) for further treatment to determine frequencies of radioactive releases to the environment.

Results of the CET analyses provide the necessary input to model and assess the transport of fission products through the drywell and containment, calculate fission product release fractions associated with containment release paths, and determine potential consequences associated with each fission product release category.

The postulated initiating events addressed in the at-power PRA are derived from a review of boiling water reactor (BWR) nuclear power plant operating experience, as summarized in NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987–1985," issued February 1999. NUREG/CR-5750 builds on previous industry studies with similar objectives, such as NUREG/CR-3862, "Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessments," issued May 1985. The NUREG/CR-5750 categories are applicable, in general, to all BWR (and pressurized-water reactor (PWR)) plants currently in operation. Some systems in the ESBWR design differ from those in the operating BWR plants. In addition, the ESBWR design contains several innovative systems; thus, certain NUREG/CR-5750 categories are not directly applicable to the ESBWR. Initiating event frequencies are estimated based on generic industry data for operating reactors, as well as on ESBWR design-specific information.

Accident sequence event tree structures and end states are defined for each initiating event category based on a review of industry PRAs and guidance documents. These are modified based on specifics of the ESBWR design and expected operation. Event tree nodal inputs are system fault tree logic or nodal point estimates, as appropriate. Functional success criteria are based on analysis of the ESBWR design and expected operation.

System fault trees were developed based on standard industry techniques and reflect the design of the ESBWR. System success criteria are based on analysis of the ESBWR design and expected operation.

Pre-initiator and post-initiator human error probabilities were defined based on the ESBWR design and expected operation. The human error probabilities used in the model are conservative screening values extracted from industry and NRC publications.

Component failure probabilities were estimated based on generic industry data and ESBWR design-specific information. CCF data derived for the ESBWR are used where available (e.g., for diesel generators, batteries, motor-operated valves (MOVs), and pumps). Generic CCF factors are used when component-specific data are not available. In order of preference, the sources used to estimate the CCF parameters are the Electric Power Research Institute (EPRI) ALWR utilities requirements document (URD), issued April 1992; NUREG/CR-5497, "Common Cause Failure Parameter Estimations," issued October 1998; and NUREG/CR-5801, "Procedures for Analysis of Common Cause Failure in Safety Analysis," issued April 1993. The methodology described in NUREG/CR-4780, "Procedures for Treating Common Cause Failures in Safety and Reliability Studies," Volume 1 (issued January 1988) and Volume 2 (issued January 1989), is applied. The multiple Greek letter (MGL) method was used to estimate the CCF probabilities.

Severe accident phenomena are explicitly addressed and are quantitatively treated. The Risk-Oriented Accident Analysis Methodology (ROAAM) is used to assess the containment response to severe accident phenomena. A linked fault tree approach is used to address the containment systems and the ability to prevent overpressurization from loss of decay heat removal.

To support the consequence analysis, multiple radionuclide release categories are modeled. Source terms are defined based on ESBWR thermal-hydraulic (T-H) analysis. Bounding consequence analyses are performed, showing that the ESBWR design meets NRC safety goals with sufficient margin.

The external events portion of the PRA explicitly analyzes core damage accidents initiated during power and shutdown operation for the following hazards:

- internal floods
- internal fires
- high winds
- seismic events

The external events analyses are bounding assessments that are meant to show significant design margin for these hazards. The frequencies of initiating events are based on generic industry data and are applied in a bounding manner. The fault trees and event trees developed for the internal events evaluations are used in the external events analyses to the maximum extent possible, using logic flags that account for the common failures induced by the external hazard events. The ESBWR seismic assessment is a seismic margin analysis (SMA). The analysis demonstrates that the ESBWR plant and equipment can withstand an earthquake with a magnitude at least 1.67 times that of the safe-shutdown earthquake (SSE).

19.1.2.1.2 Update and Maintenance of the PRA

The applicant described the PRA maintenance and update program in the design control document (DCD). The key elements of this program are summarized in this section.

The applicant treated the ESBWR PRA model as a controlled document containing the detailed information for the model. The applicant has established the following set of requirements and design controls that COL holders who reference the ESBWR design certification (DC) must implement:

- Personnel performing PRA analyses possess sufficient expertise based on training and job experience to perform the tasks.

- Personnel performing technical reviews and independent verifications of PRA analyses possess sufficient expertise based on training and job experience to perform the tasks.

- Procedures are in place that control documentation, including revisions to controlled documents and maintenance of records.

- Procedures are in place that provide for independent verifications of calculations and information used in the PRA.

For a COL holder to maintain a PRA model that reasonably reflects the as-built and as-operated characteristics of a plant that references the ESBWR DC, the applicant has established the following administrative controls:

- Monitor PRA inputs and collect new information.

- Maintain and upgrade the PRA model to be consistent with the as-built and as-operated plant.

- Ensure that PRA applications consider the cumulative impacts of pending changes

- Evaluate the impact of PRA changes on previously implemented risk-informed applications.

- Maintain configuration control of the computational methods used to support the PRA model.

- Document the PRA models and procedures that implement these controls.

An independent review of the model or model elements by a qualified reviewer or reviewers is a required part of the maintenance process. When major methodology changes or upgrades are made, outside PRA experts, such as industry peer review teams, review the PRA, and their comments are incorporated to maintain the PRA current with industry practices.

19.1.2.2 Regulatory Criteria

No specific regulatory requirements govern the quality of PRAs used to support design certification. However, Regulatory Guide (RG) 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” Revision 1, issued November 2002 and RG 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” issued January 2007, provide guidance on how to assure quality in PRA applications for commercial nuclear power facilities. The fundamental objective articulated in these documents is that the scope, technical adequacy, and level of detail of an applicant’s PRA be appropriate for the application of the PRA under consideration. To meet this objective, the staff has considered the extent to which the scope, technical adequacy, and level of detail of the applicant’s PRA support the Commission’s objectives described above which govern the treatment of severe accidents for design certification.

19.1.2.3 Staff Evaluation

The staff reviewed the quality of the ESBWR PRA by conducting its own independent evaluation of the applicant’s use of models, techniques, methodologies, assumptions, data, and computational tools, as well as evaluating the applicant’s programs and processes for ensuring quality in the PRA. As with the certification of previous advanced reactor designs (e.g., the AP1000 design), the staff’s review of the quality and completeness of the ESBWR PRA included the issuance of requests for additional information (RAIs) to the applicant, followed by the evaluation of the applicant's responses to the RAIs. The staff issued over 200 RAIs to the applicant during its review of Chapter 19 of the DCD and NEDO-33201,"ESBWR Probabilistic Risk Assessment," Revision 2, issued September 2007 (hereafter referred to as the PRA report). These RAIs The staff's review covered all aspects of the PRA model and the use of the model to assess the ESBWR, including assumptions, data, modeling, quantification, uncertainties, and sensitivity studies. The applicant has responded to the majority of these RAIs, and the staff has found the responses to be acceptable. The applicant has incorporated information provided in these request for additional information (RAI) responses into Revision 2 of the PRA report and Revision 4 of the DCD, as appropriate.

The staff used reported PRA results, as well as results of sensitivity, uncertainty, and importance analyses, to focus its review. The staff used applicable insights from previous PRA studies about key parameters and design features controlling risk in its review of the ESBWR. The staff placed a special emphasis on PRA modeling of novel, e.g., digital instrumentation and control (I&C) and passive features in the design, as well as addressing issues related to these features, such as the impact of passive system T-H uncertainties on PRA success criteria and treatment of CCFs.

19.1.2.3.1 Success Criteria and Passive System Uncertainty

The issue of T-H uncertainties arises from the passive nature of the safety-related systems used for accident mitigation. Passive safety systems rely on natural forces, such as gravity, to perform their safety functions. Such driving forces are small compared to those of pumped systems, and the uncertainty in their values, as predicted by a best-estimate T-H analysis, can be of comparable magnitude to the predicted values themselves. Therefore, some accident sequences with a frequency high enough to impact results, but not predicted to lead to core damage by a best-estimate T-H analysis, may actually lead to core damage when T-H uncertainties are considered in the PRA models.

In RAI 19.1.0-1, the staff requested that the applicant address the issue of passive system performance uncertainty and its effect on passive system success criteria. In response, the applicant provided the results of sensitivity studies which varied key T-H parameters for each of the passive systems to see the effect on success criteria for a successful event sequence following a limiting initiating event. The passive systems addressed in the studies included the gravity-driven cooling system (GDCS), the isolation condenser system (ICS), the automatic depressurization system (ADS), depressurization valves (DPVs), and the passive containment cooling system (PCCS). The studies were performed with the Modular Accident Analysis Program (MAAP) code. Table 19.1-1 summarizes the results of these studies.

Table 19.1-1 Study Results

System	Acceptance Criteria	Parameters Varied	Event	Success Criteria		
				Design Basis	**Base PRA Assumption**	**Min. Required for Success[2]**
ADS/DPV	A peak cladding temperature <2200 °F	No. of valves valve size	Medium LOCA	7 of 8 DPVs	4 of 8 DPVs	3 of 8 DPVs
GDCS	A peak cladding temperature <2200 °F	No. of valves valve size MAAP parameters	Large LOCA	7 of 8 injection valves	2 of 8 injection valves from at least 1 of 3 pools	1 of 8 injection valves from at least 1 of 3 pools
PCCS	< ultimate containment pressure	Heat Ex. Heat transfer area	Large LOCA	6 of 6 heat exchangers	4 of 6 heat exchangers	2 of 6 heat exchangers
ICS		N/A[3]	N/A	3 of 4 heat exchangers	3 of 4 heat exchangers	N/A

The applicant used the MAAP4 code to evaluate T-H success criteria. The staff is aware of T-H modeling issues with the code that could compromise its ability to confirm the validity of the PRA success criteria involving minimal sets of mitigating equipment. The applicant justified the use of the MAAP code by comparing simulations of LOCAs performed with MAAP and with the Transient Reactor Analysis Code-the General Electric version (TRACG) code. However, these benchmark calculations may not reflect T-H conditions in the reactor vessel during such accidents because the design-basis accident (DBA) analysis assumptions (i.e., the single-failure criterion) regarding availability of passive mitigating systems were applied rather than the assumptions made for the PRA which are substantially more limiting. In RAI 19.1.0-1, Supplement 1, the staff has requested that the applicant address this concern by analyzing the limiting accident scenarios assuming PRA success criteria with a code such as TRACG that is clearly capable of treating the expected T-H phenomena. Such calculations would also provide a means for adequately benchmarking the MAAP code for use in analyzing additional PRA accident sequences that may be affected by T-H uncertainties associated with passive systems.

The applicant identified the limiting accident scenarios assumed in the sensitivity studies and listed in Table 19.1-1. However, the applicant did not include enough information for the staff to understand the basis for selecting the limiting accident scenarios used to determine minimum success criteria. In RAI 19.1.0-1, the staff also requested that the applicant provide the rationale for the accident scenarios selected, including any criteria that were applied in making the selections and/or the results of any parametric studies that may have been used to identify limiting scenarios.

In a presentation on T-H uncertainty, the applicant did not describe how it selected key T-H parameters that could affect the results. Such parameters include decay heat rate, containment pressure, flow resistance in piping, heat transfer area and heat transfer coefficient in the ICS and PCCS, flow area through the break, safety relief valves (SRVs), DPVs, and check valves in the GDCS. In order to understand the uncertainty in the determination of minimal success criteria, the staff has requested in RAI 19.1.0-1 Supplement 1 that the applicant identify the key parameters and describe how each was treated in the analysis (e.g., as nominal values or

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[2] Result of sensitivity study

[3] Sensitivity not performed because design-basis criteria assumed in PRA

bounding values) and, in cases where nominal parameter values were used, discuss the impact on the results of the analyses if bounding parameter values had been used.

In addition, in the analyses, the applicant applied a limit of 2200 °F for peak cladding temperature as the acceptance criterion for avoidance of core damage. The staff finds that such a criterion is acceptable for the evaluation of PRA success criteria. However, the staff has not reviewed and approved the heat transfer, transition, and film boiling models in TRACG needed for calculating peak cladding temperature in evaluations of emergency core cooling system (ECCS) performance. In RAI 19.1.0-1, Supplement 1, the staff requested that the applicant justify the use of TRACG for modeling clad heatup and approach to thermal limits in studies of PRA success criteria.

RAI 19.1.0-1, Supplement 1, covering the four items discussed above, is being tracked as an open item.

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.1.2.3.2 Treatment of Common-Cause Failures

In the PRA, the applicant determined importance measures for common-cause basic events and found that CCF of the following components produced the highest Fussell-Vessley (FV) importance measures (largest contributors to risk) of all the common-cause events:

- control rod insertion
- actuation of check valves in the GDCS
- actuation of squib valves in the GDCS
- execution of software in the instrumentation and control (I&C) systems
- actuation of squib valves in the standby liquid control system (SLCS)

In light of these results, the applicant performed a sensitivity study in which all CCFs were eliminated. The result of this study was that the CDF decreased by three orders of magnitude, which confirms the importance of CCFs in the ESBWR design.

The staff reviewed the treatment of CCFs in each of the systems modeled in the PRA. The staff identified a number of issues related to common-cause grouping of components and CCF probabilities assumed for key components. The applicant addressed these issues in responses to a series of RAIs issued by the staff. In the responses, the applicant stated that the MGL method was used to quantify failure probabilities and reported the MGL parameters used to quantify the failure probability of each common-cause basic event. The MGL method is especially appropriate for the ESBWR PRA since systems in the ESBWR have common-cause groups with up to eight members. Use of this method in the context of the general approach for treating CCFs in NUREG/CR-4780, “Procedures for Treating Common Cause Failures in Safety Analysis,” is acceptable to the staff. The referenced methods for estimating CCF parameters are also acceptable to the staff.

Section 19.1.4.1.1.4 discusses insights associated with the sensitivity of the PRA results to changes in specific CCF probabilities.

19.1.2.3.3 Probabilistic Risk Assessment Technical Adequacy

The staff also considered the extent to which the applicant's PRA conforms to existing consensus standards for PRA (American Society of Mechanical Engineers (ASME)-RA-Sb-2005, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications" that have been endorsed by the NRC staff. The applicant has stated that "Where applicable, ASME-RA-Sb-2005 capability Category 2 attributes are included in the analysis." In RAI 19.1-117, the staff requested that the applicant (1) identify those high-level requirements or capability Category 2 attributes of the standard that have not been embodied in the ESBWR PRA, (2) address the impact on the qualitative and quantitative results of the PRA of not including those high-level requirements or capability Category 2 attributes of the standard that are applicable but have not been incorporated, and (3) describe any self-assessment or peer review process that has been performed for the ESBWR PRA and the resulting findings and observations. **RAI 19.1-155 is being tracked as an open item.**

Due to the open item that remains to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.1.2.3.4 Probabilistic Risk Assessment Maintenance and Update Program

RG 1.200 describes the elements of a PRA maintenance and update program that is acceptable to the NRC staff. The staff has reviewed the applicant's proposed program and determined that the program includes the key elements described in RG 1.200. The program described by the applicant in the DCD is therefore acceptable.

19.1.3 Special Design Features

19.1.3.1 Summary of Technical Information

19.1.3.1.1 Design/Operational Features for Preventing Core Damage

Revision 2 of the applicant's PRA report and appropriate sections of the ESBWR DCD describe the design and operational features of the ESBWR aimed at preventing core damage. These features are summarized below.

- For prevention and mitigation of an anticipated transient without scram (ATWS), the ESBWR is designed with the following features:

  – an alternate rod insertion (ARI) system that utilizes sensors and logic that are diverse and independent of the reactor protection system (RPS)

  – electrical insertion of fine motion control rod drives (FMCRDs) that also utilize sensors and logic that are diverse and independent of the RPS

  – automatic feedwater runback under conditions indicative of an ATWS

  – automatic initiation of SLCS under conditions indicative of an ATWS

  – elimination of the scram discharge volume in the control rod drive system (CRDS)

  DCD Section 15.5.4 provides details on the effectiveness of these design features for addressing ATWS concerns. Given these features, ATWS contributes insignificantly to CDF and LRF, as shown in the ESBWR PRA.

- The design of the ESBWR reduces the possibility of an intersystem loss-of-coolant accident (ISLOCA) outside containment by designing to the extent practicable all piping systems, major system components (pumps and valves), and subsystems connected to the reactor coolant pressure boundary (RCPB) to an ultimate rupture strength at least equal to the full RCPB pressure. Because of these design features of the ESBWR, ISLOCA is not a significant contributor to initiating events or accidents.

- The ESBWR design reduces the frequency and consequences of LOCAs resulting from large-diameter piping failure by removing the recirculation system altogether.

- The ICS consists of four totally independent trains, each containing an isolation condenser (IC) that condenses steam on the tube side and transfers heat to the isolation condenser/passive containment cooling system (IC/PCCS) pool, which is vented to the atmosphere. The ICs, which are connected by piping to the reactor pressure vessel (RPV), are placed at an elevation above the source of steam (i.e., vessel). When the steam is condensed, the condensate is returned to the vessel via a condensate return line. The ICS is designed as a safety-related system to remove reactor decay heat following reactor shutdown and isolation in a passive way and with minimal loss of coolant inventory from the reactor, when the normal heat removal system is unavailable following any of the following events:

  – sudden reactor isolation from power operating conditions
  – station blackout (SBO) (unavailability of all alternating current (ac) power)
  – ATWS
  – LOCA

  The ICS also prevents unnecessary reactor depressurization and operation of other engineered safety features that can also perform this function. In the event of a LOCA, the ICS provides additional liquid inventory from an inline condensate reservoir upon opening of the condensate return valves to initiate the system.

- The GDCS provides emergency core cooling passively after any event that threatens the reactor coolant inventory. Once the nuclear boiler system (NBS) has been depressurized via the automatic depressurized system (ADS), the GDCS is capable of passively injecting large volumes of water into the depressurized RPV to keep the fuel covered over both short and long timeframes following system initiation.

- The fuel and auxiliary pools cooling system (FAPCS) is designated as a backup system for low-pressure coolant injection (LPCI). In LPCI mode, the system provides makeup water from the suppression pool to the RPV through one of the main feedwater lines after the reactor has been sufficiently depressurized. The FAPCS can also provide backup shutdown cooling water. The FAPCS can provide cooling water during the long term using a pipe connection to convey water to the ICS/PCCS pool for post-LOCA heat removal after 72 hours.

- During a total loss of offsite power, the safety-related electrical distribution system is automatically powered from the onsite, non-safety-related diesel generators. If, however, these diesel generators are not available, each division of the safety-related system independently isolates itself from the non-safety-related system, and the safety-related batteries of each division provide uninterrupted power to safety-related loads of each safety-related load division. The divisional batteries are sized to provide power to required loads for 72 hours. In addition, devices that monitor the input voltage and frequency from the non-safety system and isolate the division automatically on degraded conditions protect each division of the safety-related system. The combination of these factors in the design minimizes the probability of losing electric power from onsite power supplies as a result of the loss of power from the transmission system or any disturbance of the non-safety-related ac system. Because of the nature of the passive safety-related systems in the ESBWR, SBO events are not significant contributors to CDF or LRF.

- The PCCS is a safety-related, passive-acting CHR system that maintains the containment within its design pressure and design temperature limits for design-basis accidents (DBAs) including LOCAs and post-blowdown events. The PCCS also provides a flowpath for released steam vapor back to the RPV through the GDCS. Because the PCCS is highly reliable as a result of its redundant heat exchangers and totally passive component design, the probability of a loss of CHR is significantly reduced.

- The fire protection system (FPS) serves as a preventive feature for severe accidents in two ways. First, it reduces or eliminates the possibility of damaging fire events that could induce transients, damage mitigation equipment, and hamper operator responses. Second, it supplies a means for long-term makeup to the upper containment pools, which may be required after the first 72 hours of an accident requiring passive heat removal.

19.1.3.1.2 Design/Operational Features for Mitigating the Consequences of Core Damage and Preventing Releases from Containment

Revision 2 of the applicant’s PRA report and appropriate sections of the ESBWR DCD describe the design and operational features of the ESBWR aimed at mitigating accident progression following core damage and preventing release of radioactivity from the containment. A summary follows:

- The ESBWR containment structure is designed to a higher ultimate pressure than used for currently operating BWRs. The 95-percent confidence fragility of the ESBWR primary containment system to overpressurization for the 260 °C (500 °F) steady-state thermal condition is 1.22 megapascals (MPa) (gauge) (177 pounds per square inch gauge (psig)) limited by leakage at the drywell head flange as the result of bolt yielding. Under normal operating (ambient) thermal conditions, the fragility is 1.28 MPa (gauge) (186 psig) limited by tearing of the liner at the reinforced concrete containment vessel (RCCV) wall connection with the top slab. For a 538 °C (1000 °F) transient thermal condition, the fragility is 0.89 MPa (gauge) (129 psig) limited by leakage at the bolted flange connection in the equipment hatch. The drywell head is protected from these extreme temperatures because of insulation around the RPV and restricted flowpaths from the drywell space into the area beneath the drywell head. The pool of water on top of the drywell head also keeps the flanges and closure bolts at moderate temperatures.

Within the containment is the wetwell, including the suppression pool; an upper drywell (UDW) region surrounding the RPV; and a lower drywell (LDW) region below the RPV. There are vacuum breakers between the wetwell air space and the UDW, and the UDW and LDW regions communicate freely.

- The vacuum relief function limits the magnitude of a negative pressure differential between the drywell and the suppression pool. Three drywell-to-suppression pool vacuum breakers installed in the diaphragm floor accomplish this function. These vacuum breakers operate passively in response to a negative drywell-to-suppression pool pressure gradient and are otherwise held closed by a combination of gravity and the normally positive pressure gradient.

  Four position sensors are located around the disk periphery of the primary vacuum breakers to confirm to the plant operator that the disks are securely seated. The analysis in the PRA assumes that the position switch that provides annunciation in the control room can sense a gap between the disk and the seating surface lower than 1 cm^2 (0.155 $in.^2$).

  Each vacuum breaker is equipped with a diverse, redundant, passive, process-actuated check-type isolation valve, which provides isolation capability if the vacuum breaker sticks open or leaks in its closed position. The isolation valve is normally in the closed position and, like the vacuum breaker itself, is process-actuated by differential pressure between the structure and component (SC) and drywell. In this manner, the isolation valve is more like a redundant vacuum breaker than an isolation valve, and both valves would have to leak simultaneously to create a leakage path from the SC to the drywell.

- Prevention of a combustible gas deflagration in the ESBWR containment is assured in the short term following a severe accident because the ESBWR containment is maintained in an inert condition. In the longer term, the oxygen concentration increases as a result of the continued radiolytic decomposition of the water in the containment. However, the applicant’s analysis of the ESBWR design shows that the time required for the oxygen concentration to increase to the de-inerting value of 5 percent is significantly greater than 24 hours, which allows ample time for recovery actions.

- The containment isolation system (CIS) protects against release of radioactive materials to the environment as a result of accidents occurring in systems or components within the containment. The isolation of lines and ducts that penetrate the containment boundary provides this protection. The ESBWR containment design minimizes the number of penetrations. This impacts the severe accident response because the probability of containment isolation failure is smaller.

- The probability of a high-pressure core melt is significantly reduced by the highly reliable depressurization system. The ESBWR RPV is designed with an ADS that provides automatic and effectively permanent depressurization of the reactor. In a severe accident, depressurization can prevent a high-pressure core melt ejection and the subsequent consequences. If the reactor vessel fails at an elevated pressure, fragmented core debris could be transported into the UDW. The resulting heating of the UDW could potentially pressurize and fail the drywell. Successful automatic depressurization system (ADS) actuation before vessel failure eliminates these direct containment heating (DCH) failure concerns. In addition, the following ESBWR containment design features mitigate the possible effects of high-pressure core melt:

- The containment is segregated into a UDW and an LDW, which communicate directly, but this design mitigates the ability of high-pressure core melt, ejected within the lower drywell, to reach the UDW.

- The UDW atmosphere can vent into the wetwell through a large vent area.

- The containment steel liner is structurally backed by reinforced concrete, which cannot be structurally challenged by DCH.

- The deluge mode of GDCS operation provides flow to flood the LDW when the temperature in the LDW increases enough to be indicative of RPV failure and core debris in the LDW. Of the four main deluge lines, one is available from each of the GDCS pools A and D, and two from GDCS pool BC. Each main line forks into three injection lines for a total of 12; each injection line has one squib valve. Flooding of the LDW after the introduction of core material minimizes the potential for energetic fuel-coolant interaction (FCI) at RPV failure. Covering core debris with water provides scrubbing of fission products released from the debris and cools the corium, thus limiting potential core-concrete interaction (CCI). The basemat internal melt arrest and coolability device (BiMAC) gives additional assurance of debris bed cooling by providing an engineered pathway for water flow through the debris bed.

- The BiMAC device is a passively cooled barrier to core debris on the LDW floor. This boundary is provided by a series of side-by-side inclined pipes, forming a jacket, which is passively cooled by natural circulation when subjected to thermal loading. The GDCS pools supply water to the BiMAC device via squib valves that are activated on the deluge lines. The timing and flows are such that cooling becomes available immediately upon actuation, and the chance of flooding the LDW prematurely, to the extent that this opens up a vulnerability to steam explosions, is remote. The core debris coolability analysis shows that the BiMAC device is effective in containing the potential core melt released from the RPV in a manner that assures long-term coolability and stabilization of the resulting debris.

19.1.3.1.3 Design/Operational Features for Mitigating the Consequences of Releases from Containment

Revision 2 of the applicant's PRA report and appropriate sections of the ESBWR DCD describe the design and operational features of the ESBWR aimed at mitigating the consequences of a release of radioactivity from the containment. The following describes and summarizes these features:

- The design of the ESBWR containment provides for holdup and delay of fission product release should the containment integrity be challenged. Delay in fission product release helps reduce the amount of radioactivity released and allows more time for implementation of emergency preparedness actions which lower the dose to the population.

- The deluge mode of GDCS operation provides flow through the BiMAC to flood the LDW when the temperature in the LDW increases enough to be indicative of RPV failure and core debris in the LDW. Covering core debris with water provides scrubbing of fission products released from the debris and helps reduce the magnitude of any release to the outside environment.

19.1.3.1.4 Uses of the Probabilistic Risk Assessment in the Design Process

In RAI 19.1-73, the staff requested that the applicant address uses of PRA in the design process and include a discussion of representative examples of ways in which the addition or modification of design features or operational requirements enhance the ESBWR design. The applicant provided this information in Chapter 18 of the PRA report, which is referenced in the DCD.

In its response to the staff's request, the applicant provided a list of design features that contribute to the low CDF and balanced risk profile of the ESBWR. Key examples include the following:

- The ESBWR design reduces the reliance on ac power by using 72-hour batteries for several components. A diesel-driven pump has been added as a diverse makeup capability. The core can be kept covered without any ac sources for the first 72 hours following an initiating event. This ability significantly reduces the consequences of a loss of preferred (offsite) power initiating event. These features combined with passively designed front-line safety systems eliminate SBO as a significant contributor to risk.

- ATWS events are low contributors to plant CDF because of the improved scram function and passive boron injection.

- The ESBWR design reduces the frequency and consequences of LOCAs resulting from large-diameter piping failure compared to those in BWR plants currently operating because the ESBWR design does not include a primary coolant recirculation system and its associated large-diameter piping.

- The design of the ESBWR reduces the possibility of a LOCA outside the containment because, to the extent practical, the ultimate rupture strength of all piping systems, major system components (pumps and valves), and subsystems connected to the RCPB has been set at least equal to the full reactor coolant pressure boundary (RCPB) pressure.

- The probability of a loss of CHR is significantly reduced because the PCCS is highly reliable as the result of redundant heat exchangers and a totally passive component design.

The ESBWR PRA has been used to identify and quantify various alternatives for improving the reliability of certain design features found in currently operating BWRs. For example, fire suppression piping has been rerouted based on the risk assessment results. This reduces the probability of internal flooding, which can disable multiple trains of equipment. The following are examples of PRA-based changes incorporated in the ESBWR design that have contributed to a significant improvement in plant safety:

- added redundant, physically separated flowpaths to the low-pressure injection and suppression pool cooling lines in response to fire analysis

- determined the loads to be served by the diverse protection system (DPS), which supplies diverse control signals to safety functions

- improved the design of digital controls to reduce the likelihood of inadvertent actuation of specified systems

- added redundant supply valves for ICS and PCCS pool makeup

- added redundant drainline valves for the ICS to eliminate a dependency on power supplies

- changed the routing of fire suppression piping to reduce the likelihood of room flooding

- determined the appropriate locations of control and instrumentation cabinets and power supplies to ensure physical separation

- added the BiMAC to reduce the consequences of severe accidents

19.1.3.2 Regulatory Criteria

The staff has considered the special design features of ESBWR design with respect to the Commission's objectives for new reactor designs stated in Section 19.1.1. The following two objectives are especially relevant to the evaluation of design features aimed at reducing risk:

1. Assess the balance between features of the design that prevent and mitigate accidents.

2. Determine whether the plant design represents a reduction in risk compared to the risk from existing operating plants.

19.1.3.3 Staff Evaluation

Based on the information provided by the applicant and summarized herein, it is clear that the ESBWR design includes many features that can prevent severe accidents and many that can mitigate the consequences of severe accidents. For example, the design includes features for the specific purpose of reducing the likelihood of an ATWS, loss of DHR event, core uncovering during LOCAs and ISLOCAs, as well as fires and floods. All these events have contributed significantly to risk in current operating plants and were addressed by design and operational changes after the facilities were built and operating. In addition, the ESBWR design includes features that address specific containment failure modes:

- DPVs and structural improvements to the containment which address DCH from high-pressure melt ejection

- GDCS deluge and the BiMAC which address potential melt-through of the containment

- fewer containment penetrations which reduces the likelihood of containment bypass

The staff finds that the applicant has provided an adequate balance between features of the design that prevent or mitigate accidents.

In its response to RAI 19.1-73, the applicant described the differences and similarities between the ESBWR design and the current generation of operating BWRs. It is clear from this comparison, as well as the above summary of ESBWR design features, that the ESBWR standard design has evolved from current BWR technology through the incorporation of several passive design features and other design changes intended to make the plant safer. The information provided in response to RAI 19.1-73 and summarized above indicates that the applicant has included several features in the ESBWR design to address the major contributors to core damage in current generation BWRs (i.e., SBO, ATWS, and LOCA). In addition, the ESBWR design includes features to address specific containment failure modes.

19.1.3.4 Conclusions

Based on the substantial number of design improvements in areas that have traditionally been strong contributors to risk, the staff concludes that the ESBWR design reflects a reduction in risk compared to the design of BWRs currently operating. This conclusion is consistent with the quantitative results of the ESBWR PRA which indicate a much lower total CDF and LRF compared to those of BWRs currently operating.

19.1.4 Safety Insights from the Internal Events Probabilistic Risk Assessment for Operations at Power

19.1.4.1 Results and Insights from the Level 1 Internal Events Probabilistic Risk Assessment

The staff reviewed the results of the applicant's Level 1 PRA for operations at power and found them to be mostly quantitative and lacking an adequate discussion of the following topics:

- major contributors to risk
- key qualitative risk insights for the ESBWR
- major design and operational features that contribute to reduced CDF for the ESBWR design compared to BWR plants currently operating

The applicant provided additional information on the items below in response to RAI 19.1-68 and incorporated this information into Revision 2 of the PRA report.

- discussion of key risk insights and key assumptions in the PRA model
- discussion of ESBWR design features that reduce risk
- comparison of BWR versus ESBWR PRA prevention and mitigation functions
- descriptions of the top 10 accident sequences and top 200 cutsets contributing to CDF
- results of a quantitative assessment of the risk importance of SSCs
- results and insights from 16 sensitivity studies

19.1.4.1.1 Summary of Technical Information

The applicant reports a total CDF resulting from internally generated accident sequences during power operations of 1.22×10^{-8}/yr.

The applicant identified the following key risk insights regarding the ESBWR design:

- Dominant sequences typically do not contain independent component failures. Instead, they consist of CCFs that disable entire mitigating functions. It is important to note that multiple mitigating functions must fail in the dominant sequences. A single common-cause event is not sufficient to directly result in core damage.

- The ESBWR Level 1 PRA CDF is significantly impacted if the non-safety-related systems are not credited. If the analysis takes credit for all the key backup non-safety systems, the focused Level 1 PRA results are reduced by almost 2 orders of magnitude. However, the impact to the CDF can be minimized by about an order of magnitude if the analysis credits only the availability of the DPS (including surrogate logic for DPS signal for main steam isolation valve (MSIV) isolation).

- ATWS events are low contributors to plant CDF because of the improved scram function and passive boron injection.

- In core damage sequences involving failure of the ICS where high-pressure makeup has failed and either failure to depressurize occurs or low-pressure injection is not available, the failure of the PCCS or the failure to provide makeup to the pools is not a significant contributor to CDF.

Chapter 18 of Revision 2 of the PRA report discusses additional insights.

19.1.4.1.1.1 <u>Significant Accident Sequences Leading to Core Damage.</u> Section 19.2.3.1.1 of the DCD and Chapter 7 of the PRA report describe the significant accident sequences leading to core damage. The 10 most significant sequences, which constitute approximately 80 percent of the CDF, are summarized below.

(1) An inadvertent stuck-open relief valve, where feedwater injection fails and the CRDS pumps fail to restore vessel water level and manual depressurization with SRVs fails. Automatic reactor vessel depressurization is successful with DPVs, and drywell-to-wetwell vacuum breakers suppress containment pressure. However, low-pressure injection with GDCS, FAPCS, and firewater fail. This sequence accounts for about 17 percent of the total CDF.

(2) A turbine trip transient where the control rods for reactor shutdown fail to insert, but feedwater runback is successful. SRVs lift and overpressure protection is successful. The ADS inhibit function is successful, but the SLCS fails to inject a sufficient amount of borated water. This sequence accounts for about 11 percent of the total CDF.

(3) A loss of feedwater transient where ICs fail to provide overpressure protection, and steam relief through SRVs succeeds in providing overpressure protection. Reactor vessel depressurization is successful with depressurization DPVs, and drywell-to-wetwell vacuum breakers suppress containment pressure. Low-pressure injection using GDCS, FAPCS, firewater, and CRDS pumps all fail. This sequence accounts for about 9 percent of the total CDF.

(4) An inadvertent stuck-open relief valve, where feedwater injection fails and the CRDS pumps fail to restore vessel water level. Manual depressurization with SRVs is successful, but low-pressure injection with FAPCS and firewater injection fails. Automatic reactor vessel depressurization with DPVs fails, precluding the use of the

GDCS for low-pressure injection. This sequence accounts for about 7 percent of the total CDF.

(5) A turbine trip transient where the control rods for reactor shutdown fail to insert, but feedwater runback is successful. SRVs lift and overpressure protection is successful, but one valve fails to reclose. The ADS inhibit function and the SLCS are successful. The feedwater system and the CRDS both fail to provide makeup to maintain vessel water level. This sequence accounts for about 7 percent of the total CDF.

(6) An inadvertent stuck-open relief valve, where feedwater injection fails and the CRDS pumps fail to restore vessel water level. Manual depressurization with SRVs and automatic reactor vessel depressurization with DPVs both fail, precluding the use of all low-pressure injection systems. This sequence accounts for about 6 percent of the total CDF.

(7) A loss of offsite power transient where the control rods for reactor shutdown fail to insert, but feedwater runback is successful. SRVs lift and overpressure protection is successful, but one valve fails to reclose. The ADS inhibit function and the SLCS are successful. The CRDS fails to provide makeup to maintain vessel water level. This sequence accounts for about 7 percent of the total CDF.

(8) An inadvertent stuck-open relief valve, where feedwater injection fails and the CRDS pumps fail to restore vessel water level. Manual depressurization with SRVs is successful, but low-pressure injection with FAPCS and firewater injection fails. Automatic reactor vessel depressurization with DPVs succeeds, but low-pressure injection with GDCS fails. This sequence accounts for about 5 percent of the total CDF.

(9) A large LOCA in feedwater line B depressurizes the reactor vessel. Drywell-to-wetwell vacuum breakers are successful, and pressure suppression succeeds. GDCS, FAPCS, and the firewater system all fail to provide low-pressure injection. This sequence accounts for about 4 percent of the total CDF.

(10) A loss of offsite power transient, where the ICs fail to provide overpressure protection. The SRVs lift, overpressure protection is successful, and all SRVs reclose. The ADS is successful using DPVs. Drywell-to-wetwell vacuum breakers and pressure suppression are successful. GDCS, FAPCS, and the firewater system all fail to provide low-pressure injection. This sequence accounts for about 4 percent of the total CDF.

<u>19.1.4.1.1.2 Leading Initiating Event Contributors to Core Damage from the Level 1 Internal Events Probabilistic Risk Assessment.</u> Transients contribute the most to CDF, approximately 85 percent. The most significant groups of transient initiators are as follows:

- inadvertent stuck-open relief valve (36.5 percent)
- general transients (20.4 percent)
- loss of feedwater transients (16.7 percent)
- loss of offsite power transients (11.5 percent)

LOCAs that occur inside containment contribute approximately 9 percent. The most significant LOCA initiator with respect to CDF contribution is the large steam break in feedwater line B, which represents about 4 percent of the overall CDF, thus becoming the fifth most important initiating event. Finally, breaks outside containment (BOCs) represent less than 3 percent of the total value of the CDF.

An examination of the relative contributions to the CDF of the accident classes used to define the Level 1 end states of the event trees offers another perspective on the Level 1 PRA results. Core damage events occurring at low RPV pressures with the containment initially intact account for approximately 46 percent of the CDF. Core damage events occurring at high RPV pressures with the containment initially intact account for approximately 37 percent of the CDF. Core damage events that involve a failure to insert negative reactivity (i.e., ATWS) account for about 15 percent of the CDF. Events that involve a radiological release path that bypasses the containment at the time of core damage account for less than 2 percent of the CDF.

The most significant CCFs contributing to core damage, as determined by their FV importance measures, are the following:

- failure of control rods to insert
- failure of all check valves in the GDCS
- failure of software in the instrumentation, logic, and control systems
- failure of all squib valves in the GDCS
- failure of all squib valves in the SLCS

19.1.4.1.1.3. Risk-Significant Equipment/Functions/Design Features, Phenomena/Challenges, and Human Actions

As part of its PRA, the applicant performed a study of the sensitivity of individual system failures on the results of the PRA. Based on this study, the applicant identified the following systems as the most important from a risk perspective:

- ADS
- ICS
- CRDS
- safety-related and non-safety-related control and instrumentation systems
- RPS
- GDCS

Important operator actions involve recognizing the need for depressurization or providing low-pressure injection in particular scenarios; failure to restart feedwater pumps during certain ATWS scenarios; and pre-initiator valve positioning errors in the FAPCS, control rod drive (CRD), and reactor closed cooling water (RCCW) systems. The human factors engineering program incorporates information on important operator actions.

Section 19.1.3 discusses important design features.

19.1.4.1.1.4 Insights from the Uncertainty, Importance, and Sensitivity Analyses. The applicant conducted a series of sensitivity studies on the Level 1 PRA model and stated that the purposes of these studies were to develop a better understanding and provide insights related to CDF generated through model analysis and provide guidance for ongoing design and operational activities in the consideration of overall risk impact. Table 19.1-2 summarizes these studies and their key results.

Table 19.1-2 Sensitivity Studies and Key Results

Sensitivity Study	Description	Impact on CDF[4]
Human Reliability	**Description**	Significant change
Common Cause Failure	All CCF eliminated	1000-fold decrease
Test and Maintenance (T&M) Unavailability	All T&M activities failures; increase unavailability by factor of 10	Small increase; negligible increase
SLCS Success Criteria	One train for success instead of two	Small decrease
Component Type Code Data	Basic event data for six component groups increased by factor of 10	Little or no change
SRV Common-Cause Factors	One common-cause group versus one for each of the two valve functions (ADS and overpressure protection)	No change
SPC & LPCI Success Criteria	Two trains for success instead of one	Small increase
Turbine Bypass Valve Success Criteria	6 of 12 versus 4 of 12 valves for success	Negligible
LOCA Frequency	All frequencies doubled	Small increase
LOCA-IC Frequency	Frequency of LOCAs outside containment increased to reflect more piping outside	No change
CRD Injection Post-Containment-Failure	CRD assumed to fail if containment fails	No change
Accumulators	All accumulators supporting pneumatic components failed	100-fold increase
Vacuum Breakers	Failure rate increased by factor of 10	Small increase
System Importance	Importance measures computed for 31 systems	N/A
Demand for Passive Systems	CDF for sequences having passive component failure compared to CDF for sequences having passive component success	Sequences involving failure of ICS components are large fraction of CDF

Key insights derived from these studies are presented below:

- Sensitivity study results indicate that changes in the human error failure probabilities, particularly preinitiators, have the potential to impact CDF.

- Sensitivity study results indicate that squib valve failure rate estimates have the potential to impact CDF.

[4] This assessment is provided by the applicant.

- Sensitivity study results indicate that the preinitiator operator actions have a significant impact on the risk achievement worth (RAW) value. This is primarily because of the many potential latent failures and relatively high reliability estimate for each of these operator actions.

- Accident sequences in which DPVs are challenged contribute to approximately 61 percent of the CDF. In two-thirds of the cases, the DPVs are demanded and are successful; in one-third of the cases, the DPVs are demanded and fail.

- The PRA model conservatively assumes that a single failure on either train of the SLCS causes core damage if the control rods fail to insert (ATWS). The CDF would be reduced by approximately 13 percent if either train of the SLCS is able to mitigate ATWS scenarios.

- Changes to the squib valve failure rate have a significant impact on CDF. Increases in the failure rate of the squib valves used for the ADS cause a significant increase in the accident Class III contribution (core damage at high reactor vessel pressure). Similarly, an increase in the failure rate of the squib valves in the GDCS causes a significant increase in the accident Class I contribution (core damage at low reactor vessel pressure). However, an increase only to the SLCS squib valves does not have a very pronounced impact on CDF.

- An increase of 1 order of magnitude of the vacuum breaker and backup valve failure rate causes the CDF to increase only by approximately 10 percent.

19.1.4.1.2 Regulatory Criteria

The staff has considered the results and insights from the Level 1 PRA with respect to the Commission’s objectives for new reactor designs stated in Section 19.1.1. The following four objectives for the applicant’s use of the design PRA are especially relevant to the evaluation of results and insights from the Level 1 PRA:

(1) Reduce or eliminate the significant risk contributors of existing operating plants that are applicable to the new design by introducing appropriate features and requirements.

(2) Identify risk-informed safety insights based on systematic evaluations of the risk associated with the design such that the applicant can identify and describe (a) the design’s robustness, levels of defense-in-depth, and tolerance of severe accidents initiated by either internal or external events and (b) the risk-significance of specific human errors associated with the design.

(3) Determine how the risk associated with the design compares against the Commission’s goal of less than $1x10^{-4}$/yr for CDF.

(4) Determine whether the plant design represents a reduction in risk compared to existing operating plants.

19.1.4.1.3 Staff Evaluation

The applicant has reported a CDF of 1.22×10^{-8}/yr for internal events initiated during power operation. In contrast, comparable CDFs for the majority of existing BWR operating plants reported in the individual plant examination (IPE) program (see NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," issued October 1997) are between 1×10^{-6}/yr and 1×10^{-4}/yr. This difference in CDFs reflects the differences in design between currently operating BWRs and the ESBWR, as discussed below.

In NUREG-1560, which reports the results of the IPE program, the staff identified CDFs for the major initiating event categories and design features and human actions that had a significant impact on the contribution of those events to the CDF. The comparison of these design features and human events with the ESBWR design in Table 19.1-3 provides insight regarding the difference in CDFs.

Table 19.1-3 Comparison of Design Features in Existing BWRs and the ESBWR by Event Category

Event Category	Design Features in Existing BWRs That Significantly Affect CDF (NUREG-1560)	Relevant ESBWR Design Features
SBO	• availability of cooling systems that are independent of ac power, battery life, and overall reliability of ac and dc power systems (reduces CDF)	During a total loss of offsite power at an ESBWR-based plant, the safety-related electrical distribution system is automatically powered from the onsite non-safety-related diesel generators. If these diesel generators are not available, then each division of the safety-related system independently isolates itself from the non-safety-related system, and the safety-related batteries of each division provide power to safety-related loads of each safety-related load division. The divisional batteries are sized to provide power to required loads for 72 hours. In addition, the ESBWR design includes safety-related DHR systems that do not require ac power to operate. Consequently, SBO events are not significant contributors to CDF for the ESBWR.
Transients with Loss of Injection Capability	• degree of dependency of injection systems on support systems; low dependency reduces CDF	The ESBWR design includes a large number of injection systems (i.e., GDCS, CRD, FAPCS, and the fire water system). In addition, the GDCS is designed to run with no dependency on support systems for the first 72 hours following an accident. Also, injection into the reactor vessel with a diesel-driven fire pump is part of the ESBWR design which is unlike current operating plants.

Event Category	Design Features in Existing BWRs That Significantly Affect CDF (NUREG-1560)	Relevant ESBWR Design Features
Transients with Loss of DHR Capability	• degree of dependency of DHR systems on support systems; low dependency reduces CDF • the capability of the emergency core cooling systems (ECCSs) to pump saturated water; reduces CDF • use of reactor water cleanup system as an alternative DHR system; reduces CDF • ability to replenish water sources outside containment for use in long-term cooling	DHR systems in the ESBWR include the ICS and the PCCS which are passive systems designed to run with no dependency on support systems for the first 72 hours following an accident. ECCSs in the ESBWR are gravity driven and do not rely on pumps. Adequate cooling water inventory is guaranteed for 72 hours, and after that, makeup is provided, by design, using the diesel-driven fire pump.
LOCA	high redundancy and diversity in injection systems; reduces CDF	The ESBWR design has high redundancy and diversity in injection systems. It includes passive injection systems, motor-operated active injection systems, and diesel-driven injection systems.
ATWS	reliance on success of human actions; increases CDF	In the ESBWR, some human actions have been automated (e.g., automatic initiation of SLCS). In addition, the ESBWR adds several important ATWS mitigation features, including the ARI system, FMCRD insertion capability, automatic feedwater runback, and elimination of the scram discharge volume.

As described in Table 19.1-3, the ESBWR includes a number of new design features and design modifications which specifically address issues important to risk in previous BWR designs. It is reasonable to expect, based on these changes, that the CDF would be substantially lower than the CDF for currently operating plants. However, some of these features and changes rely on new technology with uncertain reliability (e.g., squib valves in passive systems and digital I&C systems). The applicant has addressed this by examining the sensitivity of the CDF to changes in reliability data for these features or by choosing data believed to be conservative or bounding and by examining the impact of uncertainty in passive system success criteria on CDF. Table 19.1-2 summarizes sensitivity studies involving squib valve failure rate and CCF data. Section 19.1.2.3.1 documents the staff's evaluation of passive system success criteria.

19.1.4.1.4 Conclusions

The staff has reviewed the results and insights derived from the Level I PRA and sensitivity studies. Based on this review, the staff concludes that the applicant has performed adequate systematic evaluations of the risk associated with the design and used them to identify risk-informed safety insights in a manner consistent with the Commission's stated goals.

The staff has considered the reported CDF for the ESBWR baseline PRA in relation to CDFs reported for currently operating BWRs, the risk-significant design differences between the

ESBWR and currently operating BWRs, and the applicant's studies of the sensitivity of the computed CDF to changes in modeling and data in the PRA. Based on these considerations, the staff concludes that the methodology and results of the Level 1 risk analysis described in the ESBWR PRA are acceptable and meet the Commission's goal of less than $1x10^{-4}$/yr for core damage. The staff concludes that the ESBWR design represents a reduction in risk compared to existing operating BWR plants.

As discussed above, the applicant has incorporated substantial features into the ESBWR design specifically aimed at reducing the risk from SBO and LOCA events. As a result, the staff concludes that the applicant has reduced significant risk contributors of existing operating plants that are applicable to the new design, by introducing appropriate features and requirements, consistent with the Commission's stated goals.

19.1.4.2 Results and Insights from the Level 2 Internal Events Probabilistic Risk Assessment (Containment Analysis)

The following sections present results and insights from the Level 2 portion of the ESBWR full-power internal events PRA. These sections address the frequency of the various accident classes considered in the Level 2 analysis, the frequency and conditional containment failure probability (CCFP), a breakdown of containment failure frequency in terms of important containment failure/release modes, and a summary of the risk-significant insights from the Level 2 PRA and the supporting sensitivity analyses.

19.1.4.2.1 Summary of Technical Information

The ESBWR has a very low LRF ($9.6x10^{-10}$/yr), and accident sequences leading to such results are not only unlikely but also have broad bands of uncertainties associated with them. Consequently, the applicant used a bounding approach, rather than a best-estimate method, for assessing containment performance. The applicant also estimated that the ESBWR passive containment design is sufficiently robust to effectively mitigate the consequences of severe accidents with a low attendant CCFP of 0.08.

The applicant identified the following key insights relevant to preventing or mitigating large releases to the environment:

- The containment provides a highly reliable barrier to the release of fission products after a severe accident, with the dominant release category being that defined by technical specification leakage (TSL).

- The ESBWR is designed to minimize the effects of direct containment heat, ex-vessel steam explosions (EVEs), and CCI. Its containment is designed to a higher ultimate pressure than that of conventional BWRs.

The applicant also stated that, given a severe accident, venting would occur when the containment pressure reaches 90 percent of the ultimate containment strength.

19.1.4.2.1.1 Level 2 Probabilistic Risk Assessment Methodology

The Level 2 PRA analysis focuses on the response of the containment and its systems during the progression of severe accidents. The methodology used includes binning the Level 1 PRA results into a manageable number of accident classes and constructing and quantifying containment event trees (CETs), simulating severe accident progression and containment challenges for a number of accident sequences that represent the significant core damage scenarios, and assigning representative sequence results into release categories for the purpose of defining the end states and determining the pathways of radioisotopes into the environment. The containment response has been evaluated for a 24-hour period following the onset of core damage. The CCFP is determined from the Level 2 PRA.

Results of the CET analyses provide the necessary input to model and assess fission product transport through the containment, calculate radiological release fractions associated with containment release paths, and determine potential consequences associated with each fission product release category.

The Level 1 PRA results are grouped into a set of classes for transfer into the containment evaluation. The results of the containment evaluation are then grouped into a set of “release categories” for use as source terms for the offsite consequence analysis and, subsequently, risk integration.

A Level 2 PRA quantification model was created with the same basic methodology as the Level 1 model. In the Level 2 model, the initiator is actually a gate under which the appropriate Level 1 sequences are binned. Effectively, the integrated model is a combination of both the Level 1 and Level 2 PRA models. As such, all initiator impact is preserved throughout the quantification, and no special treatment is required for scenarios such as loss of preferred power (LOPP). In each of the Level 2 CETs, the nodes are modeled as either a fault tree to represent system functions or a basic event with a point estimate to represent phenomenological effects.

The fault trees may be completely independent of Level 1 sequences (such as the GDCS deluge system) or contain dependencies (such as short-term CHR). Integrating the Level 2 PRA with the Level 1 as a single, one-time quantification model allows all dependencies and initiator impacts to be correctly reflected in the results.

19.1.4.2.1.1.1 Containment Event Trees

To determine the conditional system failure probabilities values used on the CET branches, the 156 listed Level 1 quantified accident sequences above the cutoff level of $1.0\text{x}10^{-15}$/yr are sorted into subclasses based on the Level 1 accident class binning and the water level in the LDW at the time of vessel breach (to determine the fraction of sequences that are susceptible to EVE).

The Level 1 accident classes, discussed in Section 7 of the PRA, are as follows:

- Class I: Vessel failure occurs at low pressure (<1 MPa). (46 percent of CDF)
- Class II: Containment failure precedes core damage. (0.3 percent of CDF)
- Class III: Vessel failure occurs at high pressure (>1 MPa). (37 percent of CDF)
- Class IV: Vessel failure occurs at low pressure; core damage results from failure to insert negative reactivity in ATWS conditions. (15 percent of CDF)

- Class V: Core damage occurs with the RPV open to the environment because of BOCs. (1 percent of CDF)

As shown in Table 19.1-4, a set of rules based on break size, location, and injection status is used to bin the low-pressure Class I and Class IV sequences into three subgroups according to the water level existing in the LDW at vessel breach. If the water is above 1.5 meters, the applicant conservatively assumes that the pedestal fails as the result of steam explosion. If the water level is between 0.7 and 1.5 meters, a steam explosion is possible, but failure of the pedestal is physically unreasonable (PU). If the water level is below 0.7 meters, the applicant has determined that a steam explosion impulse would not challenge the containment structure.

The CETs were used to evaluate the complete spectrum of potential challenges to containment integrity. They address both containment system functions relevant to mitigating the overpressure and bypass challenges and phenomenological effects. The Level 1 sequence bins were used as the initiators, or entry events, to the CETs, which were constructed using point estimates for phenomenological effects and appropriate logic to account for mitigating system success or failure by establishing the logically possible containment responses. Finally, the end states of the CETs, which are termed “release categories,” were defined. Release categories represent meaningfully different outcomes to the containment challenge and were used in the source term evaluation.

Table 19.1-4 Assignment of Level 1 Accident Sequences to Level 2 Containment Event Tree Entry Events

Level 1 Accident Class	Class CDF (per year)	Class Summary	LDW Water Level Bin	Level 2 CET Entry Event	CET-Assigned CDF (per year)	CDF Fraction
Class I	5.6×10^{-9}	Sequences with RPV failure at low pressure	Low/Dry	I_LD	5.0×10^{-9}	0.40
			Medium	I_M	1.0×10^{-10}	0.01
			High	I_H	5.9×10^{-10}	0.05
Class II	4.2×10^{-11}	Containment failure preceding core damage	No CET required as the containment is failed in these sequences before core damage			
Class III	4.5×10^{-9}	Sequences with RPV failure at high pressure	Low/Dry	III_LD	4.5×10^{-9}	0.37
Class IV	1.9×10^{-9}	Sequences involving failure to insert negative reactivity	Low/Dry	IV_LD	1.9×10^{-9}	0.15
			Medium	IV_M	1.0×10^{-12}	<0.01
			High	IV_H	2.2×10^{-11}	<0.01
Class V	1.5×10^{-10}	Breaks outside of containment	No CET required as there is direct communication between the RPV and the environment			

The seven CET entry events are associated with the accident classes as shown in Table 19.1-4. The event trees include top events, depending on the class, that address the following:

- Phenomena
  - DCH
  - EVE
  - dry and wet molten core coolant interaction (MCCI)
  - core debris cooling
- System Functions
  - CIS
  - GDCS deluge function
  - Vapor suppression function

- CHR, short term
- CHR, long term
- Actuation of containment venting

Either a (phenomenological) basic event with an assigned point value or a (system) fault tree represents each of the CET nodes. Section 21 of the PRA describes the treatment of the phenomenological events by the ROAAM procedure. The events addressed include containment performance against DCH, containment and BiMAC performance against EVE, and containment and BiMAC performance against basemat melt penetration (BMP) and overpressurization from gases produced from core debris-concrete interactions.

A complete Level 2 fault tree analysis was done for the GDCS deluge system. Because the deluge system is completely independent of all other plant systems, it is also independent of all Level 1 sequences.

Conditional (depending on initiator effects and Level 1 sequences) probabilities for the failure branches of the other system functions in the CETs are calculated by means of the Level 2 fault trees developed for these nodes.

19.1.4.2.1.1.2 Simulation of Accident Progression and Containment Challenges

As discussed above, the Level 1 analysis grouped severe accidents into five categories. With the exception of Class V accidents, in which the containment is completely bypassed, a single dominant sequence was selected to represent each of the accident classes for detailed modeling. In this way, the containment response to the complete spectrum of accidents contributing to the CDF could be evaluated.

Table 19.1-5 (adapted from Table 8.3-1 of the PRA) identifies the sequences used to represent each accident class. The “core damage sequence descriptor” used in the table derives from the results of the Level 1 analysis. The core damage descriptor key (used in Tables 19.1-5 and 19.1-6) is as follows:

Key:

MLi: medium liquid break (injection line)
T: transient
T-AT: transient without negative reactivity insertion
nCHR: no CHR
nDP: no depressurization
nIN: no injection
FR: filtered release (controlled vent)
TSL: technical specification leakage
NA: not applicable

The representative sequences are based on the Level 1 results presented in Section 7 of the PRA and the definitions of the Level 1 sequence bins. For example, Table 7.2-3 of the PRA indicates that about 80 percent of the Class I frequency is associated with stuck-open relief valve (T-IORV), large feedwater LOCA (LL-S-FDWA/B), or loss of feedwater (T-FDW) sequences. From the perspective of modeling the containment response to a severe accident, all Class I sequences can be represented as a transient with loss of injection T_nIN and successful depressurization. A similar approach was used in selecting the representative sequences for the other accident classes.

Table 19.1-5 Representative Core Damage Sequences

Accident Class	Core Damage Sequence Descriptor	Sequence Summary
I	T_nIN	Transient initiator followed by no short- or long-term coolant injection. ADS functions. ICS not credited. PCCS available, but no active CHR (FAPCS). GDCS/BiMAC function successful.
II	MLi_nCHR	Medium liquid line break: GDCS injection line. System is depressurized and injection systems function. CHR not available.
III	T_nDP_nIN	Transient initiator followed by no short- or long-term coolant injection. The RPV is not depressurized; pressure controlled at relief valve setpoint. ICS not credited. PCCS available, but no active CHR (FAPCS). GDCS/BiMAC function successful.
IV	T-AT_nIN	Transient followed by failure to insert negative reactivity. ICS not credited. RPV is not initially depressurized (ADS inhibit successful). SLCS is ineffective or unavailable. Feedwater runback is successful. No short- or long-term coolant injection. PCCS available, but no active CHR (FAPCS). GDCS/BiMAC function successful.
V	None	No representative sequence assigned for containment evaluation as Class V events involve direct communication between the RPV and environment.

Table 8.3-2 of the PRA couples each representative core damage sequence with various release categories and their associated frequencies. The resulting scenarios are assigned containment response sequence descriptors to summarize the core damage and containment release information, thus providing additional information by presenting the release category frequency in terms of the contribution from each accident class.

To determine the key characteristics of the containment response to a severe accident, an ESBWR simulation model was developed using MAAP, Version 4.0.6. MAAP includes models for the important phenomena that might occur in a severe LWR accident. The model offers insights into the timing of severe accident progression, the containment pressure-temperature response, and ultimately the potential source term if the containment were to fail. The source term calculations support the characterization of the timing and release magnitude of the release categories, which is used as input to the Level 3 PRA calculations. Table 19.1-6 shows the results of MAAP simulations of the ESBWR representative sequences. Appendix 8B to the PRA shows graphs of many additional representative sequence results, including pressures, temperatures, water levels, and hydrogen concentrations, to provide complete documentation of the containment analysis.

The applicant did not use MAAP to estimate the probability of containment failure from DCH, EVE, or basemat penetration events caused by BiMAC failure. Instead, the applicant used the ROAAM procedure, as reported in Section 21 of the PRA.

Table 19.1-6 Summary of Results of Severe Accident Sequence Analysis

Sequence Descriptor	RPV Depress. Initiated (seconds)	Core Uncovered (hours)	Onset of Core Damage (hours)	RPV Failure (hours)	Deluge Actuated (hours)	Drywell Pressure 24 hours after Onset of Core Damage (MPa)
T_nIN _TSL	665	0.50	1.1	7.8	7.8	0.58
T_nIN_nCHR_FR	661	0.48	1.3	7.7	7.7	0.92
MLi_nCHR	124	>72	>72	NA	NA	NA
T_nDP_nIN_TSL	NA	0.92	1.7	6.2	6.2	0.57
T_nDP_nIN_nCHR _FR	NA	0.93	1.7	6.7	6.7	1.01
T-AT_nIN_TSL	1163	0.1	0.1	5.9	6.0	0.57
T-AT_nIN_nCHR_FR	1161	0.1	0.1	5.7	5.7	1.1

Accident Class I involves sequences where the RPV fails at low pressure; this accident class represents approximately 46 percent of the CDF. Accident Class III involves sequences in which the RPV fails at high pressure; this accident class represents approximately 37 percent of the CDF. Accident Class IV includes sequences that are initiated by an ATWS and followed by failure to initiate negative reactivity. Such sequences represent approximately 15.6 percent of the CDF. Transient sequences in which there is no core injection dominate all three classes. The sequences T_nIN, T_nDP_nIN, and T-AT_nIN were used to evaluate the containment response to Class I, III, and IV events, respectively.

Accident Class II involves sequences in which containment failure precedes RPV failure. After containment failure, RPV makeup capability is assumed to be lost because the gradual boil-off of water in the passive systems may result in damage to piping connections which would render active makeup systems unavailable. As a result, core damage and RPV failure occur after containment failure. As shown in representative sequence MLi_nCHR, core damage does not occur during the first 72 hours after the accident.

Sequence T_nIN_TSL (Represents Class I)

The T_nIN sequence simulates a transient initiated by an LOPP where no short- or long-term coolant injection to the RPV by the feedwater system, CRDS, or GDCS is available. The ADS functions to reduce the RPV pressure. Heat removal by the ICs is not credited because of the low reactor pressure. Short-term CHR is accomplished by successful PCCS functioning; PCCS pool makeup is successful, thus allowing long-term CHR. The GDCS deluge system and BiMAC are available for debris bed cooling. With successful containment isolation, vapor suppression, and CHR, the containment remains intact. Most likely, TSL is the mode of fission product release.

In this event, the primary system experiences delayed depressurization because of the opening of the first ADS-actuated valves at about 655 seconds. The pressure in the containment increases as the drywell is filled with steam and heats up. The core becomes uncovered about 30 minutes into the event. Fuel rod heatup and fission product release, hydrogen production from oxidation of the fuel cladding, and fuel melting follows. The fission products and hydrogen are swept into the containment through the DPVs as the core melts. This leads to further heating and pressurization of the drywell air space.

The RPV lower head penetrations fail about 7.8 hours into the event. Core debris is deposited on the LDW floor, leading to a temperature increase high enough to cause the GDCS deluge line to open. The GDCS pool water then drains into the LDW and covers the debris bed. The BiMAC functions as designed to quench the debris, so that, significant CCI does not occur. Therefore, no significant fission product aerosols or noncondensable gases are generated in the ex-vessel phase of the accident sequence.

The core debris in the LDW heats the water pool, generating steam that pressurizes the containment until the PCCS heat removal capacity becomes consistent and comparable to the decay heat generated by the core debris. The containment pressure reaches about 0.58 MPa 24 hours after the onset of core damage and before the time when containment venting would be implemented. Radionuclide release to the environment occurs only through potential containment leakage as the containment remains intact and venting is not required.

Sequence MLi_nCHR (Represents Class II)

The initiating event for the sequence MLi_nCHR is a medium LOCA, assumed to occur in the GDCS injection line. Failure of CHR is followed by containment pressurization to its ultimate capacity. Core cooling occurs by gravity feed through the GDCS injection and equalizing lines. Eventually, the water used for RPV makeup is boiled off.

The containment pressurizes until the ultimate strength is reached at about 31 hours. The ADS depressurizes the RPV, which allows GDCS tanks to drain into the RPV and then into the LDW through the break. The shroud water level initially rises in response to the GDCS tank injection, then decays as the GDCS inventory is depleted. The shroud level decreases below the elevation of the break at about 6 hours. Further shroud level decrease occurs until flow through the equalizing line begins at about 8.3 hours. Flow from the suppression pool maintains the RPV level above the top of active fuel (TAF) for about 71 hours. Shortly thereafter, core heatup begins.

The results of the sequence simulation indicate that the core damage following containment failure as the result of loss of CHR does not occur within a 24-hour period after accident initiation. In fact, core temperatures do not reach the point of fuel damage until more than 72 hours after accident initiation. Given the long time for mitigating action to supplement RPV makeup, Class II events are not considered contributors to the offsite consequence analysis.

Sequence T_nDP_nIN_nCHR_FR (Represents Class III)

The initiating event for the sequence T_nDP_nIN is a loss-of-offsite power. This sequence differs from T_nIN in that depressurization fails, although the SRVs remain functional in the relief mode. The ICS is not credited. The CRD and feedwater systems are unavailable. The RPV fails at about 6.2 hours, with the RPV at a pressure close to the relief valve setpoint.

Actuation of the GDCS deluge line and successful BiMAC function prevent significant core debris-concrete interaction (CCI) from occurring in the LDW. Material dispersed to the UDW does not result in significant CCI because the large dispersal area allows the material to be cooled. Continued heating of the water by debris in the LDW leads to continued steam generation, which increases containment pressure. The PCCS removes heat from the containment, thus preventing overpressurization.

For the case where CHR has failed, the containment pressure increases, and controlled venting is implemented to limit the pressure rise and control the radionuclide release point. The drywell pressure has reached 0.62 MPa (less than 70 percent of the ultimate containment strength) 24 hours after onset of core damage; thus, venting would not likely be implemented in this timeframe. The 90-percent assumption is met at 42.5 hours after accident initiation, which is about 2.9 hours before containment overpressurization would occur.

Sequence T-AT_nIN_TSL (Represents Class IV)

Sequence T-AT_nIN is a general transient followed by an ATWS. The SLCS is ineffective or unavailable. The RPV is not initially depressurized because the ADS is successfully inhibited. To control the ATWS power level, feedwater runback is successful with operator control assumed at the TAF. The PCCS is available, but no active CHR (e.g. FAPCS) is assumed.

Control of core water level just above the TAF results in a core power level of about 30 percent of full power 3 minutes after the transient begins. At that time, it is assumed that feedwater is terminated and safety system injection to the RPV does not occur. (System pressure prevents gravity drain from the GDCS, and the CRDS is unavailable for forced flow.) Because the ADS inhibit is successful, the RPV is maintained at high pressure, controlled by the SRV setpoint, until the core water level decreases below the point of effective cooling. At that point, manual depressurization is initiated, but injection into the RPV continues to be unsuccessful. RPV failure occurs at about 5.9 hours at low pressure.

Similar to sequence T_nDP_nIN_nCHR_FR, CCI is minimal in both the LDW and UDW regions, because the GDCS line is actuated and the BiMAC functions as designed. The PCCS removes heat from the containment, thus preventing overpressurization.

The containment pressure reaches about 0.57 MPa 24 hours after onset of core damage, well below the point at which containment venting would be implemented. Radionuclide release to the environment occurs only through potential containment leakage as the containment remains intact and venting is not required.

For all of the representative sequences, the containment is intact at 24 hours, and no fission product releases have occurred by this time.

19.1.4.2.1.1.3 Release Category Definitions

The containment response to a severe accident is depicted by the end states of CETs. These end states become the release categories that are used to characterize potential source terms. The source terms are used in the offsite consequence analysis.

Each end state of the CET set is assigned to 1 of 11 containment release categories. Of the release categories, 10 are containment failure or bypass modes. If no containment failure or bypass occurs, the release associated with allowable technical specification leakage is assumed. Table 19.1-7 summarizes the release categories.

Table 19.1-7 Release Categories, End States, and Release Paths

Release Category	End-State Description	Significant Factors	Release Path
BOC	Unisolated piping break outside of containment	Feedwater, main steam, RWCU/SDC line breaks	RPV to environment
BYP	Loss of isolation	CIS function failure	Drywell to environment
CCID	LDW corium debris not flooded; CCI noncondensable gas ruptures drywell	GDCS deluge is unsuccessful	Drywell to environment
CCIW	LDW corium debris bed flooded but not effectively cooled; CCI gas ruptures drywell	GDCS deluge is successful	Drywell to environment
DCH	DCH event (RPV failure at high pressure) overpressure ruptures drywell or fails liner	Physically unreasonable. No failure is assumed	Drywell to environment
EVE	EVE at RPV failure ruptures drywell	Gravity core drop into deep (>1.5 m) water pool	Drywell to environment
FR	Wetwell airspace vented before steam overpressure ruptures drywell	Suppression pool vent opened by operator	Filtered though pool
OPVB	Vacuum breakers fail to close or are open; steam overpressure ruptures drywell	Containment pressure suppression function fails	Drywell to environment
OPW1	CHR fails in first 24 hours; steam overpressure ruptures drywell	PCCS or pool cooling system failure	Drywell to environment
OPW2	CHR fails after 24 hours; steam overpressure ruptures drywell	PCCS unavailable after 24 hours	Drywell to environment
TSL	Allowed leakage from the drywell at the TSL (0.5 percent of containment air volume per day at rated pressure)	Preexisting small leak paths from drywell	Drywell to environment

19.1.4.2.1.1.4 ESBWR Conditional Containment Failure Probability

The CET quantification resulted in a cumulative containment failure frequency of 9.6 x 10^{10}/yr. The Level 1 CDF is 1.22×10^{-8}/yr, so that the ESBWR conditional containment failure probability for all non-TSL failure modes is 0.08 (the ratio of these two numbers), within the NRC's CPG of 0.10.

19.1.4.2.1.1.5 Source Term Evaluation

The source term evaluation was performed with the MAAP computer code, which produces the distribution of radionuclides released to the environment as a function of time. The source terms are input from the Level 2 PRA to the Level 3 consequence analyses.

Each release category is represented by one or two severe accident sequences selected and modeled to represent the group of potential severe accidents that could be associated with that release category. In some cases, both low-pressure and high-pressure classes were selected for the same release category to represent broader and more thorough contributions of accident sequences. For each source term, the timing, energy, isotopic content, and magnitude of release are established based on plant-specific T-H calculations using the MAAP code.

Conservative assumptions were typically made to account for analytical and phenomenological uncertainties.

The core loading inventory assumed in developing the source term is bounding for enrichment and exposure for GE14 fuel. It assumes an end-of-cycle equilibrium inventory, with a core average exposure of 36 gigawatts/metric ton of uranium (GW/MTU), a maximum discharge exposure of 58 gigawatt-days/metric ton of uranium (GWd/MTU), and a power density of 5.75 megawatt-thermal (MWt)/bundle. These are representative of the expected ESBWR operating conditions.

19.1.4.2.1.2. Significant Accident Sequences and Accident Classes Contributing to Containment Failure

Most of the release categories listed in Table 19.1-7 are associated with overpressurizing the containment. Also included are TSL, venting from the wetwell airspace in such a way as to enable fission product scrubbing by the suppression pool (FR for filtered release), failure to isolate the containment (BYP), and an unisolated pipe BOC.

Section 21 of the PRA discusses the potential for containment failure as the result of DCH, EVE, and basemat penetration in the ROAAM evaluation. Section 9 of the PRA discusses containment overpressure failure as a consequence of system failures. These failure modes, as pertinent to the ESBWR, are briefly explained below.

19.1.4.2.1.2.1 Containment Failure from Direct Containment Heating (DCH)

DCH may occur when high-velocity steam impinges on melt already released into a containment compartment, which creates regions of fine scale mixing, a large interfacial area for heat transfer, and oxidation of metallic components in the melt. In the ESBWR, the mixing occurs in the LDW, while the main receiving volume, in which de-entrainment occurs, is the UDW.

The ROAAM analysis shows that the ESBWR containment can withstand bounding DCH pressure loads and concludes that catastrophic containment failure as the result of DCH is PU.

The applicant states that the following factors support this conclusion:

- The UDW atmosphere can vent into the wetwell through a large vent area and an effective heat sink.

- The drywell head is (externally) immersed in water and essentially isolated from the UDW atmosphere.

- The containment steel liner is structurally backed by reinforced concrete, which cannot be structurally challenged by DCH.

Therefore, DCH is not a containment rupture failure mode in the PRA.

However, the calculations also show short periods of potentially very high temperatures in the LDW atmosphere (up to 4000 K). These high temperatures and the presence of potentially large quantities of melt in the LDW indicate that the LDW liner could be subject to local failures. The applicant's position is that liner failure in the LDW space would not constitute containment failure because of the presence of structural “lips” that isolate the gap space from that of the upper portions of the containment wall.

19.1.4.2.1.2.2 Containment Failure and BiMAC Failure Resulting from Ex-Vessel Steam Explosions (EVE)

EVEs are energetic FCIs that are triggered from melt-coolant mixtures that develop as the melt released from the RPV falls into and traverses the depth of a water pool below. Metallic melts such as those expected here for low-pressure scenarios are especially prone to such energetic behavior. When large quantities of melt are involved with highly subcooled water, the result is pressure pulses that are potentially capable of loading major structures to failure.

The relevant structures are the reactor pedestal (a 2.5-meter reinforced concrete wall) and the BiMAC device, a layer of thick-walled steel pipes that are well embedded in reinforced concrete in such a way that they are supported in all directions. Failure of the reactor pedestal, along with the steel liner on it, would constitute violation of the containment boundary. While the load-bearing capacity of this structure is 2.85 MPa, explosive-level pressures acting on a time scale of milliseconds can produce concrete cracking, along with liner stretching and tearing, sufficient to compromise the leak-tightness of the containment. Failure of the BiMAC device, on the other hand, is defined as crushing (or locally collapsing) the pipes so that they cannot perform their heat removal function of channeling the so-generated two-phase mixture from the bottom onto the top of the debris mass. Such failure would raise the possibility of continuing corium-concrete interactions, basemat penetration, and containment pressurization by the so-generated noncondensable gases.

The ROAAM assessment finds that failure of the ESBWR containment liner (and therefore, the leak-tightness of the containment) because of EVE is PU for shallow, saturated pools. For accidents involving deep (greater than 1.5 meters), subcooled water pools, the PRA utilizes an appropriately conservative position that, because “integrity of both the liner and the concrete structure could be possibly compromised,” the containment will rupture at RPV failure from overpressure. A sensitivity study shows that medium-depth pools are of negligible importance. On the other hand, the applicant argued that the BiMAC can resist higher dynamic loads than can the pedestal and the containment liner and therefore is not susceptible to failure as the result of EVE.

Analyses reported in Section 21 of the PRA support the conclusion that for all but 1 percent of the CDF (that is, accidents involving deep, subcooled water pools), violation of the ESBWR containment leak-tightness and of the BiMAC function as the result of EVE is PU. The applicant cites the following features to support this conclusion:

- an accident management strategy and related hardware features that prohibit large amounts of cold water from entering the LDW before RPV breach
- the physical fact that premixtures in saturated water pools become highly voided and thus unable to support the escalation of natural triggers to thermal detonations
- reactor pedestal and BiMAC structural designs capable of resisting explosion load impulses of over about 500 kilopascal-seconds (kPa-s) and about 100 kPa-s, respectively.

Similar calculations performed under NRC sponsorship (ERI/NRC-06-2062, Reference 19-40) confirm the applicant's conclusions.

A consequence of this analysis is that the ESBWR PRA assumes that an EVE adequate to fail containment occurs with a probability of 1.0 every time the core melts through the RPV and falls under gravity into an LDW with a "high" water level. The ROAAM assessment is that if the water level is "medium" or "low/dry," a sufficiently energetic steam explosion is PU. For PRA sensitivity study purposes, a failure probability of $1.0x10^{-3}$ is assigned to cases involving a medium water level in the LDW.

19.14.2.1.2.3 Containment Failure from Molten Core-Concrete Interactions (CCID and CCIW)

Section 21 of the PRA states that the BiMAC device is effective in containing all potential core melt releases from the RPV in a manner that ensures long-term coolability and stabilization of the resulting debris. There would be no significant ablation of concrete in the basemat or pedestal wall, nor would containment overpressurization by concrete decomposition gases occur. The applicant stated that the following features support this conclusion:

- A refractory ceramic material serves as a protective layer to eliminate ablation by superheated metallic jets. The layer thickness is chosen to prevent the ablation front from reaching the BiMAC pipes even under large-volume pours of superheated melts.

- Proper positioning and dimensioning of the BiMAC pipes allow for stable, low-pressure-loss and natural circulation that is not susceptible to local burnout resulting from thermal loads exceeding the critical heat flux (CHF) or to dry-outs resulting from flow- and water-deficient regimes.

- The BiMAC in the LDW can be sized and positioned in such a way that all melt released from the vessel (except of course any melt dispersed to the UDW in high-pressure scenarios) is captured and contained.

- The provision of an angle of inclination of the lower boundary can balance the various requirements, including operational space available and good margins to local burnout.

The applicant has assigned a nodal value of 0.01 for debris cooling failure following successful deluge operation based on the design of the BiMAC device.

Sequences that successfully supply water to the BiMAC but have unsuccessful BiMAC function are terminated with the release category core-concrete interactions-wet (CCIW). The category assignment indicates that the corium debris bed is successfully covered with water, but CCI proceeds because of inadequate cooling to terminate the interactions. That is, the debris bed becomes relatively impermeable to water, or for some other reason, the overlying water pool does not prevent MCCI. Systems considered in the CET will not mitigate the containment pressure rise attributable to noncondensable gas generation, which will lead to eventual containment overpressurization failure.

Sequences in which no water is supplied to the BiMAC terminate with the end-state core-concrete interactions-dry (CCID). In such sequences, the CCI would be greater than the CCIW end state because there is no debris bed cooling. High levels of aerosols and noncondensable gases are produced and eventually lead to containment overpressurization failure.

In response to RAI 19.2-32, GE Hitachi Nuclear Energy, (GEH) provided the results of sensitivity studies using MAAP, performed to estimate concrete ablation for both limestone and basaltic concrete to assess the potential for RPV pedestal failure. These cases involved a loss of injection with successful depressurization of the RPV.

The thickness of the ESBWR LDW wall (RPV pedestal) is 2.5 meters, and the thickness of the ESBWR basemat is 5.1 meters. The BiMAC, which is 1.6 meters thick, is located on top of the basemat. Breach of the pedestal would occur at an ablation depth of 2.5 meters, with a possible loss of structural integrity at a lesser depth.

The calculated times after RPV failure to horizontal ablation of 2.5 meters ranged from 26 hours (dry LDW basaltic) to 55 hours (dry LDW limestone), to beyond the 72-hour run time (limestone and basaltic in flooded LDW). An independent assessment of CCI using MELCOR 1.8.6 (ERI/NRC-07-201) confirmed that concrete ablation depths in the axial direction would be of similar or somewhat smaller magnitude than those predicted by MAAP for several comparable sequences involving assumed basaltic concrete under both dry and wet conditions. A representative MAAP calculation for CCID in Appendix 9A to the PRA shows that the containment overpressure failure limit is reached at about 20 hours after RPV failure for a basaltic concrete basemat, well before pedestal failure would occur. While it is possible that a horizontal “blowout” may occur into the lower reactor building somewhat before the 20 hours because of local thinning of the pressure boundary in the region of the BiMAC trough, further analysis of this event is of questionable value given the very low probability of a CCID-type event. It is reasonable to assume that the containment would fail from overpressurization before basemat melt-through or pedestal failure.

Assuming the successful operation of the deluge system, no credit for operation of the BiMAC, and the anticipated heat transfer to water above the debris pool, the expected response is ablation of less than half the pedestal thickness.

19.1.4.2.1.2.4 Containment Isolation System Failure (BYP)

These are events where the containment has been bypassed because of the failure of the CIS. As a result, there is a direct path from the containment atmosphere to the environment from the start of the accident (i.e., BYP).

19.1.4.2.1.2.5 Containment Heat Removal Function Failure (OPW1 and OPW2)

This is the condition where the vapor suppression capability has functioned, but there is a failure to remove heat from the containment. The containment fails by overpressurization from stored energy and decay heat. Short-term (defined as OPW1) and long-term (defined as OPW2) containment failure modes correspond to failures within 24 hours and after 24 hours of core damage, respectively.

19.1.4.2.1.2.6 Vacuum Breaker Failure (OPVB)

This is the condition where a vacuum breaker is open or fails to reclose, thus defeating the vapor suppression function, which, in turn, also fails CHR. The containment fails by overpressurization, most likely sooner than in cases represented by OPW1 and OPW2.

19.1.4.2.1.2.7 Containment Venting (FR)

The ESBWR contains a manually initiated vent connecting the suppression chamber gas space to the environment. Venting is potentially effective only in the case of CHR function failure and would serve to convert the uncontrolled overpressurization containment failure into a controlled venting path from the drywell atmosphere through the suppression pool into the environment (i.e., containment venting, referred to as FR). Forcing the radionuclide pathway to go through the suppression pool effects a filtering action. The expected operator guidance is to open the vent lines as needed to control pressure rise to less than an as yet unspecified fraction (expected to be on the order of 90 percent) of the containment ultimate pressure capacity.

19.1.4.2.1.2.8 Break Outside of Containment (BOC)

In this event, a piping BOC occurs in which the RPV communicates directly with the environment. A representative event is a reactor water cleanup (RWCU) large line break above the core, which represents a potential path from the RPV directly to the environment and a large source term.

In response to RAI 19.2-38, the applicant discussed another possible break. The analysis of a BOC in the ICS, as an initiator, shows that the break does not contribute to CDF.

Containment bypass because of an IC tube failure is not probable. A temperature-induced IC tube failure requires that the level in the IC pool be lowered as the result of boiling that uncovers the IC heat exchanger(s). The IC heat exchanger is designed to withstand the design temperature and pressure of the RPV. The IC heat exchanger will not see higher pressures without multiple failures of SRVs to control RPV pressure. Temperatures above the design temperature require that the core is first uncovered, as steam exiting the core would be at saturation temperature.

Water hammer is not probable as the IC heat exchangers are normally pressurized because of the open steam supply valves. Condensate fills the piping from the IC heat exchanger to the condensate return valves. A loop seal between the condensate return valves and the RPV is designed to ensure that steam continues to enter the IC heat exchanger preferentially through the steam riser, irrespective of the water level inside the reactor, and does not move counter-current back up the condensate return line.

The RWCU break outside containment analyzed in the PRA bounds the consequences of an IC tube failure. The RWCU BOC sequence is an unisolated BOC in the shutdown cooling piping followed by no injection into the RPV. In this scenario, the release begins at the onset of fuel damage and proceeds directly to the environment.

The release in the IC tube failure sequence would occur after fuel damage, as heatup of the uncovered IC heat exchanger is required. This sequence is a Class III sequence (core damage with the RPV at high pressure) and also requires a failure to isolate the lines.

19.1.4.2.1.2.9 Technical Specification Leakage (TSL)

The technical specifications limit allowable containment leakage to 0.5 percent of containment air volume per day at rated design-basis pressure. Leakage at the TS limit is included in all of the modeled severe accident sequences and represents the no containment failure/bypass outcome. The leakage path is conservatively assumed to occur directly between the drywell atmosphere and environment, thus bypassing the suppression pool and the reactor building heating, ventilation, and air conditioning (HVAC) system mitigation pathways.

19.1.4.2.1.3. Leading Contributors to Containment Failure from Level 2 Internal Events Probabilistic Risk Assessment

Table 19.1-8 provides the list of release categories and their contributions to containment failure. This table also shows representative cesium iodide (CsI) release fractions at 24 hours after core melt. The BOC frequency is directly calculated from failures of containment isolation for pipe breaks outside containment in the Level 1 PRA. In addition, since on the average the containment would be deinerted for a period of 24 hours per year, and containment combustible gas deflagration cannot be excluded when the containment atmosphere is deinerted, it was conservatively assumed that all core damage events during the deinerting would lead to containment failure. This contribution was calculated as $3.3x10^{-11}$(i.e., CDF/365) and added to the BYP frequency.

Table 19.1-8 Release Category Frequencies and Representative Release Fractions

Release Category	Frequency (/RY) (% contribution to Containment Failure (CF))	Representative CsI Release Fraction at 24 Hours
TSL (no CF)	$1.12x10^{-8}$ (0)	0.00016
EVE	$6.10x10^{-10}$ (63.5)	0.028
BOC	$1.47x10^{-10}$ (15.3)	0.41
CCIW	$9.90x10^{-11}$ (10.3)	0.0006
BYP	$5.60x10^{-11}$ (5.8)	0.038
OPW1	$3.20x10^{-11}$ (3.3)	0.0
OPVB	$1.60x10^{-11}$ (1.7)	0.005
CCID	$1.00x10^{-12}$ (0.1)	0.019
FR	$<1x10^{-12}$	0.0
OPW2	$<1x10^{-12}$	0.0
DCH	0	-

The applicant provided the release category frequencies for external events in an amended response to RAI 19.1-13, Supplement 1. The applicant also gave the CDFs for shutdown events. They are $9.37x10^{-9}$/reactor-year (RY), $2.71x10^{-8}$/RY, and $5.24x10^{-9}$/RY for internal events, fire, and flood, respectively. The analyses conservatively assume that these core damage scenarios result in large releases, since the containment is open during most of the shutdown period.

The quantification resulted in a summed (all release categories except for TSL) containment failure frequency of 9.6×10^{10}/yr. These are all termed “large releases.”

The low-pressure Class I accident sequences contribute the majority (73 percent) of the containment failures, almost entirely in the release category EVE. The necessary and sufficient condition for an EVE is a low-pressure RPV breach at a time when the LDW water depth is more than 1.5 meters. The dominant Level 1 sequence meeting this condition is a large LOCA with pressure suppression success and failure to inject.

Class II contributes 4 percent of the containment failures, mostly as the release category OPW1.

The high-pressure Class III contributes about 5 percent of the containment failures, almost entirely as the release category CCIW.

Class IV (ATWS type) contributes 7 percent to the containment failure probability, with equal probabilities for BYP, CCIW, and EVE.

Class V contributes 11 percent to the containment failure probability entirely as the release category BOC.

19.1.4.2.1.4. Risk-Significant Equipment/Functions/Design Features, Phenomena/Challenges, and Human Actions

Important insights from the Level 2 PRA are summarized below. These are organized in terms of equipment/design features, severe accident phenomena/challenges, and human actions.

The analysis evaluated the potential for at-power internal events containment failure as the result of combustible gas generation, containment bypass, and overpressurization. In addition, the analysis determined the frequency of containment failure events resulting from the phenomenological events discussed in Section 21 of the PRA (core debris-concrete interaction, DCH, and EVE).

Because of the ESBWR design and reliability of containment systems, the most likely containment response to a severe accident is associated with successful containment isolation, successful vapor suppression, and successful CHR. As a result, the containment provides a highly reliable barrier to the release of fission products after a severe accident, with only 8 percent of the core damage accidents resulting in releases larger than those associated with the minimal release leakage at the TS limit. This result meets the recommended goal of 10 percent.

A containment penetration screening evaluation indicated that only a few penetrations required isolation to prevent significant offsite consequences. The probability of the bypass failure mode is dominated by common-cause hardware failures, resulting in a calculated frequency of containment bypass about 3 orders of magnitude lower than the TSL release category.

19.1.4.2.1.4.1 Equipment/Design Features

The ESBWR features an inert containment atmosphere to prevent deflagration or detonation of combustible mixtures and a manually operated containment overpressurization protection system to guard against slow buildup of pressure resulting from noncondensable gas generation and/or heatup of the suppression pool water. Unlike the advanced BWR (ABWR), or any other previous GEH BWR, the ESBWR containment design includes the PCCS, to remove decay heat from the containment, and the (also passive) BiMAC device which is intended to essentially eliminate the possibility of extended corium-melt interactions, noncondensable gas generation, and basemat penetration.

Table 19.1-9 summarizes the containment challenges and mitigative attributes in place for the ESBWR. These attributes have contributed to reducing and/or eliminating the likelihood of the associated severe accident challenges (i.e., those identified as risk contributors at some of the existing BWR plants) in the ESBWR.

Table 19.1-9 Summary of Containment Challenges and Mitigative Attributes in Place

Challenge	Failure Mode	Mitigation
DCH	Energetic Drywell Failure	Pressure Suppression Vents Reinforced Concrete Support
	UDW Liner Thermal Failure	Liner Anchoring System
	LDW Liner Thermal Failure	Reinforced Concrete Barrier Cap Separation from UDW
EVE	Pedestal/Liner Failure	Dimensions and Reinforcement
	BiMAC Failure	Pipe Size and Thickness Pipes Embedded in Concrete
BMP & CCI	BiMAC Activation Failure	Sensing & Actuation Instrumentation Diverse/Passive Valve Action
	Local Melt-Through	Refractory Protective Layer

19.1.4.2.1.4.2 Phenomena/Challenges

Given a severe accident, the applicant has considered the following challenges to containment integrity:

- prompt, energetic loading: explosive FCIs, high-pressure melt ejection leading to DCH (and pressurization)

- late, gradual loading: melt ablation and penetration of the containment basemat, pressurization of containment atmosphere by steam and/or noncondensable gases

- isolation failure: errors or malfunctions that leave existing flowpaths open to the outside, activation of the containment overpressure protection system

The phenomenological (physics) components of these threats (namely, EVE, DCH, and BMP) are discussed as part of the ROAAM process (Section 21 of the ESBWR PRA). The discussion of BMP also provides the principal phenomenological input needed to assess containment overpressurization, which, because it is a systems-driven event, is treated in the Level 2 PRA. This is the case for isolation failure as well.

The applicant's ROAAM process found that for all but a very low fraction of the CDF (that is, accidents involving deep, subcooled water pools; see Section 19.1.4.2.1.3), violation of the ESBWR containment leak-tightness and of the BiMAC function as the result of EVE is PU. Also the process determined that the ESBWR containment can withstand bounding DCH pressure loads and that catastrophic containment failure as the result of DCH is PU. The staff concurs with this determination.

The applicant also found that the BiMAC device is effective in containing all potential core melt releases from the RPV in a manner that assures long-term coolability and stabilization of the resulting debris. The mode and location of lower head failure is treated as a splinter set of scenarios. A high/side failure (i.e., at some elevation above the very bottom of the RPV) would make all events more benign than in the bounding ROAAM analysis. This is because the quantities and rates of melt location from the RPV into the LDW would be significantly lower. In particular, this would tend to eliminate the DCH and steam explosion threats and would make all BiMAC-related performance even more reliable.

External events and shutdowns do not impact the accident progression or source term magnitude. They may, however, lead to failures of support systems. External event severe accidents have no direct impact on the probability of containment failure. Shutdown event analyses conservatively assume that these core damage scenarios result in large releases since the containment is open during most of the shutdown windows.

19.1.4.2.1.4.3 Human Actions

Because of the passive nature of the ESBWR containment systems, no operator actions are required to support the containment response to a severe accident in the 24-hour period after onset of core damage.

The CIS, vacuum breakers, and PCCS do not require operator action to initiate or function. Operator action is not required to maintain CHR through the PCCS for the 24-hour period after onset of core damage, and containment venting will not be required during that period.

Therefore, operator actions are considered in the containment evaluation only in the following cases:

- Action is taken as a backup to an automatic action (e.g., to open the connecting valve for PCCS pool makeup if the low-water-level signals were to fail).

- Action is taken to initiate a backup system (e.g., to actuate the FAPCS if the PCCS were unavailable).

- Actions require a long time to initiate. For example, the suppression chamber vent is under operator control. In virtually all scenarios, there would be a long period (more than 24 hours) to initiate venting to prevent containment overpressure resulting from a loss of CHR. In fact, manual actuation is desirable because the time for venting can be based on plant, weather, and evacuation information available to the operators.

Because these operator actions are redundant to passive system functioning or are required only after a long time period, such actions do not have a significant impact on the probability of containment failure.

19.1.4.2.1.5 Insights from Uncertainty, Importance, and Sensitivity Analyses

19.1.4.2.1.5.1 Uncertainty Analysis

GEH does not consider a formal uncertainty analysis to be necessary for the Level 2 portion of the ESBWR PRA because of the bounding nature of the ROAAM process for developing the CET split fractions. In these cases, the high confidence values are used rather than the mean values.

Severe accident phenomena are complex, and the details of many processes are not fully understood. One feature of the ROAAM approach is its attempt to identify areas of uncertainty, while making best use of current understanding (supplemented by experimental and analytical efforts) to allow issue closure without the need to address all details of all processes (e.g., those leading to the spontaneous triggering of a steam explosion).

The applicant acknowledges that, in ROAAM, when the basis of evaluation is epistemic, probabilities are subjective. Therefore, a numerical probability scale can be used only for the purpose of propagating uncertainties. This approach was used in all previous applications of ROAAM (as enumerated in Section 21.2 of the PRA), and the staff finds such a qualitative interpretation of the end results to be appropriate and sufficient. Application of this procedure to the ESBWR is simpler than previous applications, and the results are more robust in two ways. First, for all potential containment challenges, strongly bounding arguments can be made at a level of generality and margins that obviate the need for propagation of uncertainties. Second, according to the ROAAM "quality of evaluation criteria" (see Table 21.6-2 in the PRA), all assessments can be made independently of scenario details.

Uncertainties remain in the Level 2 PRA even given the bounding nature of the ROAAM process. Though numerical nodal failure values (branch probabilities) were assigned (typically a value of 1×10^{-3} is assigned for phenomena), no source term uncertainties in terms of time, quantity, and chemical and physical forms of release were analyzed. The ROAAM process does not cover the systems portions of the CETs, nor does it consider the propagation of the driving Level 1 PRA numbers. The Level 1 uncertainty analysis presented in Section 11 of the PRA demonstrates that the ratio of the total CDF upper bound (95th percentile) to the mean value is approximately a factor of 3. Simply assuming that the Level 2 PRA is linearly driven by this uncertainty, the ratio of the upper bound release frequency to the mean result would be similarly a factor of 3.

Nevertheless, it is believed that the bounding nature of the ROAAM process, coupled with the very low levels of CDF, containment failure probability (CFP), and the absolute risk of core damage and fission product release, does mean that uncertainty analyses for the Level 2 PRA would not produce additional insights.

19.1.4.2.1.5.2 Importance Analysis

GEH does not report results of any importance analysis for the Level 2 PRA..

19.1.4.2.1.5.3 Sensitivity Analysis

Tables 11.3-18 and 11.3-19 of Chapter 11 of the PRA report the results of three Level 2 sensitivity studies. These are summarized below. Level 3 studies, discussed in Section 19.1.4.3.5 below, address the sensitivity of offsite consequences to meteorological conditions, release elevation, release energy (heat and buoyancy), and mission time.

The Level 2 PRA generally utilizes the metric "non-TSL" (nTSL) release as the equivalent of CDF in the Level 1 model; nTSL is assumed to be equivalent to the LRF.

19.1.4.2.1.5.3.1 CIS Node Placement in the CET

A Level 2 PRA model sensitivity analysis was performed to study the effect of moving the CIS node to the first position in the event trees and to assess the impact on LRF. The current Level 2 PRA model is based on event trees with CIS in a nodal position of 3 or 4.

Results for the CIS node sensitivity analysis showed no impact to LRF as demonstrated by no change in nTSL frequency over the PRA Level 2 base model. The placement of the CIS node earlier in the event trees was shown to have little impact on the nTSL frequencies.

19.1.4.2.1.5.3.2 Physically Unreasonable Phenomenology

A current Level 2 PRA model contains containment failure modes that are considered "physically unreasonable." A sensitivity study was performed to better understand the impact to nTSL and source terms pertaining to the omission of these PU modes from the model. These modes include EVE from a medium LDW water level and DCH.

Results for the PU sensitivity analysis showed only a small increase in the nTSL frequency over the PRA Level 2 base model. A release frequency for DCH of 2.56×10^{-12} was obtained for the PU sensitivity contributing 0.2 percent to the total non-TSL release frequency. The non-DCH release category source terms were minimally affected by the increased leakage area in their respective sequences. The DCH release category itself has a high release fraction, but its low frequency renders potential offsite consequences negligible. The PU sensitivity confirms that no potentially significant offsite consequences are being negated by the exclusion of PU events from the Level 2 PRA model.

19.1.4.2.1.5.3.3 Vacuum Breakers Data

In the vacuum breaker sensitivity analysis, the failure rates of the vacuum breakers were increased by a factor of 10 in the database file to account for uncertainty in general reliability and anticipated number of cycles in the mission time.

Results for the vacuum breaker sensitivity showed an nTSL frequency of 2.13×10^{-9} at a truncation of 1×10^{-15}. This value for nTSL represents a significant increase in nTSL frequency of more than double that of the base Level 2 model. However, the increased nTSL meets the NRC goal of 1×10^{-6} /yr for LRF. Based on these results, the uncertainties associated with the primary vacuum breaker design and anticipated number of cycles may contribute to slightly increased LRF, but the increase is reasonable.

19.1.4.2.1.5.3.4 BiMAC Failure

Given failure of BiMAC and continued corium-concrete interaction, there is a potential for RPV pedestal failure. Sensitivity studies using MAAP were performed to estimate concrete ablation for both limestone and basaltic concrete. These cases involved a loss of injection with successful depressurization of the RPV. The results are discussed in Section 19.1.4.2.1.2.3 above, in the subsection titled "Core-Concrete Interaction (CCIW and CCID)."

The applicant does not consider it useful to perform LRF-based sensitivities for operator actions. First, because the total CDF estimated in this sensitivity is less than 1.0×10^{-6}/yr, it is not possible to raise the LRF value above the goal. Second, no important operator actions are credited in the LRF evaluation. For example, removing the containment vent from the LRF calculation would not affect the results because both the success of the vent and the failure of containment as a result of overpressure are treated as large releases.

19.1.4.2.2 Regulatory Criteria

The staff has considered the results and insights from the Level 2 PRA with respect to the Commission's objectives for new reactor designs stated in Section 19.1.1. The following five objectives for the applicant's use of the design PRA are especially relevant to the evaluation of results and insights from the Level 2 PRA:

(1) Reduce or eliminate the significant risk contributors of existing operating plants that are applicable to the new design by introducing appropriate features and requirements.

(2) Identify risk-informed safety insights based on systematic evaluations of the risk associated with the design such that the applicant can identify and describe the design's robustness, levels of defense-in-depth, and tolerance of severe accidents initiated by internal events.

(3) Determine how the risk associated with the design compares against the Commission's goals of less than $1x10^{-6}$/yr for LRF. In addition, compare the design against the Commission's approved use of a CPG, which includes (a) a deterministic goal that containment integrity be maintained for approximately 24 hours following the onset of core damage for the more likely severe accident challenges and (2) a probabilistic goal that the CCFP be less than approximately 0.1 for the composite of all core damage sequences assessed in the PRA.

(4) Assess the balance between features of the design that prevent and mitigate accidents.

(5) Determine whether the plant design represents a reduction in risk compared to existing operating plants.

19.1.4.2.3 Staff Evaluation

An extensive compilation of the results generated by the industry in performing its IPEs for the current generation of plants is published as NUREG-1560. The staff's observations on BWR containment performance include the following:

- The large-volume containments of PWRs are, on average, less likely to experience early structural failures than the smaller BWR pressure suppression containments.

- Overpressure failures, primarily from ATWS, FCI, and failures resulting from direct impingement of core debris are important contributors to early failure for most BWR containments; hydrogen burns are important in some Mark III containments.

- The higher probability of early structural failures of BWR Mark I plants, compared to the later BWR containments, is driven largely by drywell shell melt-through.

- Bypass is generally not important for BWRs.

- Overpressurization when CHR is lost is the primary cause of late failure in most PWR and some BWR containments.

- High-pressure and temperature loads caused by CCIs are important for late failure in BWR containments.

- Some Mark I IPEs have found that containment venting is important for avoiding late uncontrolled failure.

The staff's review of Chapter 19 of the ESBWR DCD and Sections 8–11 and 21 of the PRA verifies that the ESBWR design is more robust and has greater tolerance for severe accidents than that of the operating plants. Specific findings include the following:

- The LRF for internal events is calculated by the applicant to be 9.6×10^{-10}/yr, and the CCFP is calculated to be 0.08. The LRF is more than 3 orders of magnitude below the Commission's safety goal, and the CCFP is acceptably low. This is a significant reduction in risk as compared with existing BWRs, which typically have LRF values in the range of 1.0×10^{-6}/yr to 1.0×10^{-5}/yr and CCFPs up to 0.7, with an average value around 0.3.

- The design features and requirements introduced by the applicant reduce or eliminate significant risk contributors identified in existing operating plants. These features provide a good balance between prevention and mitigation.

  – The new features designed to prevent or mitigate ATWS greatly reduce the probability and/or consequences of ATWS and hence LRF.

  – Designing all piping systems, pumps, valves, and subsystems connected to the RCPB to an ultimate strength equal to or greater than the full RCPB pressure is a preventive measure that reduces the likelihood of ISLOCA and consequent containment bypass probability and hence LRF.

  – Since the ESBWR containment is designed to a higher ultimate pressure than that of currently operating BWRs, there is a higher likelihood of averting containment failure and hence a reduction in LRF and CCFP. The containment would be more likely to survive for at least 24 hours following the onset of core damage.

  – The probability of a high-pressure core melt is reduced as a result of a highly reliable ADS. This system plays a role both in preventing and mitigating severe accidents. It reduces the likelihood of early containment failure from DCH. Moreover, drywell segregation into upper and lower regions, and the ability to vent the UDW atmosphere into the wetwell through a large venting area, would mitigate the effects of a high-pressure core melt. Consequently, the risk impacts of high-pressure core melt events (LRF and CCFP) are reduced in comparison to those of current-generation BWRs.

  – The deluge mode of GDCS operation, in concert with the BiMAC device, would act to further reduce the likelihood of containment failure, either from overpressurization, drywell liner melt-through, or from basemat penetration from core debris attack. Moreover, the design procedure of not immediately adding water greatly reduces the probability of a highly energetic steam explosion. Consequently, LRF and CCFP are further reduced relative to current-generation BWRs.

– The wetwell vent is available to avert catastrophic containment failure. It would not be needed during the first 24 hours after core damage and would be opened only if the containment pressure exceeded 90 percent of its ultimate capacity.

An independent assessment of the ESBWR design response to selected severe accident scenarios was performed for the NRC using the latest version of the MELCOR 1.8.6 computer code (ERI/NRC-07-201, Reference 19-41). The assessment examined 13 accident scenarios from the ESBWR PRA, which were chosen based on a combination of frequency, consequence, and/or dominant risk. The majority of these scenarios were similar or identical to sequences analyzed with MAAP4 by GEH in Revisions 1 and 2 of the PRA, and the assessment compared the results of corresponding sequences and release categories in the two studies. The results generally support and confirm the PRA accident progression analysis methodology and the GEH interpretations of its analyses of the ESBWR reactor, containment, and system response to severe accidents. With respect to the predicted radiological source terms, differences were observed for some release categories and fission product classes between the MELCOR 1.8.6 and MAAP4 results, in particular for FR and late containment overpressure (OPW2). However, these two release categories are minor contributors to the ESBWR overall severe accident risk as determined by the PRA. For most release categories and fission product classes, MELCOR 1.8.6 and MAAP4 results either closely agree or differ by an amount that is within the margin attributable to fission product transport and other modeling uncertainties and to possible differences in scenario boundary conditions. Therefore, in the area of radiological release, the independent assessment using MELCOR 1.8.6 is viewed as generally supporting the results and conclusions of the source term analysis conducted in the ESBWR PRA.

19.1.4.3 Results and Insights from Level 3 Internal Events Probabilistic Risk Assessment

The Level 3 PRA is performed to assess the calculated ESBWR public risk level results to three major offsite consequence-related goals established in the GEH ESBWR licensing review bases based on the NRC Safety Goal Policy Statement.

The intent of the following implemented design goals is to ensure that the ESBWR design takes the approach of keeping radiological risk as low as reasonably achievable:

(1) Individual Risk Goal

NRC: The risk to an average individual, within 1.6 kilometers (1 mile) of the plant site boundary, of prompt fatalities that might result from reactor accidents should not exceed 0.1 percent of the sum of “prompt fatality risks” resulting from other accidents to which members of the U.S. population are generally exposed. For this evaluation, the sum of prompt fatality risks is taken as the U.S. accidental death risk value of 39.1 deaths per 100,000 people per year.

GEH: As a design objective, the individual risk goal is conservatively set to be 3.9×10^{-8} fatalities per year within 1.6 kilometers (1 mile), which is 1 order of magnitude lower than the evaluated NRC safety goal

(2) Societal Risk Goal

NRC: The risk to the population, in the area within 16.1 kilometers (10 miles) of a nuclear power plant, of cancer fatalities that might result from nuclear power plant operation should not exceed 0.1 percent of the sum of the “cancer fatality risks” resulting from all other causes. The cancer fatality risk is taken as 169 deaths per 100,000 people per year.

GEH: Similar to the individual risk goal, the design objective sets the societal risk goal at 10 percent of the NRC goal, or 1.7×10^{-7} fatalities per year within 16.1 kilometers (10 miles).

(3) Radiation Dose Goal

NRC: The probability of exceeding a whole body dose of 0.25 sievert (Sv) at a distance of 805 meters (.5 miles) from the reactor shall be less than $1.0 \times 10\text{-}^{6}$/yr.

GEH: The design objective for the probability of receiving 0.25 Sv at 0.5 mile is set at less than 1.0×10^{-7}/yr, which is an order of magnitude lower than the NRC’s dose goal.

The staff agrees that these constitute a reasonable set of goals to establish the level of public risk for the ESBWR and that they are consistent with the Safety Goal Policy Statement.

19.1.4.3.1 <u>Level 3 Probabilistic Risk Assessment Methodology</u>

The Level 3 PRA defined risk in terms of person-rem and calculated it by multiplying the yearly frequency of an event by its consequences. The consequences were defined as the committed effective dose equivalent (50-year committed) to the total population within a 10- and a 50-mile radius of the plant. The MELCOR Accident Consequence Code System (MACCS2), Version 1.13, was used to estimate accident consequences. The MACCS2 code evaluates offsite dose and consequences such as early fatality risk and latent cancer fatality risk for each source term (i.e., radionuclide release category) over a range of possible weather conditions and evacuation assumptions. The calculated results are compared to consequence-related goals to determine if the goals are satisfied. Effective doses were estimated for each of 10 different release categories.

For the ESBWR Level 3 PRA, each of the 10 non-zero frequency release categories is represented by one or two severe accident sequences that were selected and modeled to represent the group of potential severe accidents that could be associated with that release category. In some cases, both low-pressure and high-pressure classes were selected for the same release category to represent a broader and more thorough contribution of accident sequences. For each source term, the timing, energy, isotopic content, and magnitude of release are established based on plant-specific, T-H calculations using the MAAP code.

This analysis uses a meteorological condition comparable to the EPRI ALWR URD meteorological reference data set, which is indicated to be a meteorological data set significantly worse than conditions at the average U.S. site. The SANDIA siting study population density data are used to develop a uniform population density. A bounding uniform density of 305 people per km^2 (790 people per mi^2) for the first 32 kilometers (20 miles) is used for all radial intervals. The evacuation parameters used in this analysis are termed conservative assumptions in that no evacuation or relocation in terms of physical movement is assumed and no sheltering is assumed. The public is assumed to continue normal activity during the reactor accident in this bounding analysis. For baseline analysis, each release category and associated source term is modeled to occur at ground level. The thermal content of the plume is assumed to be the same as ambient.

The staff finds the overall approach to consequence analysis and the use of the MACCS2 code to be consistent with the present state of knowledge regarding severe accident modeling and therefore acceptable.

19.1.4.3.1.1 Results

Table 19.1-10 summarizes the baseline results for internal events occurring during full-power operation and compares them to the evaluated NRC safety goals.

Table 19.1-10 Baseline Consequence Goals and Results (from PRA Rev. 2, Table 10.4-2)

Goal	**RISK GOAL**	**ESBWR Risk @24 Hours†**	**Safety Goal Achieved @24 Hours†**	**ESBWR Risk @72 Hours†**	**Safety Goal Achieved @72 Hours†**
Individual Risk (0–1 mi)	$<3.9\times10^{-7}$ (0.1%)	7.4×10^{-11}	YES	8.2×10^{-11}	YES
Societal Risk (0–10 mi)	$<1.7\times10^{-6}$ (0.1%)	9.0×10^{-12}	YES	1.1×10^{-11}	YES
Probability of Radiation Dose >0.20 Sv (at 0.5 mi)	$<10^{-6}$	2.03×10^{-9}	YES	2.10×10^{-9}	YES

† After the onset of core damage

Risk and consequence results in terms of the safety goals are not available for external events and shutdown modes. The staff has asked the applicant for these data in RAI 19.1-13, Supplement 1. **RAI 19.1-13, Supplement 1, is being tracked as an open item.**

Surrogate values are available in the PRA for all but seismic events. Sections 19.1.5 and 19.1.6 list external event and shutdown CDF and LRF results. The values listed are of the same magnitude as those for the at-power internal events case. Seismic events are not expected to add to the risk significantly, based on the seismic margin study results. Because the individual CDF values are developed with differing levels of conservatism, the applicant indicated that it is not meaningful to add CDF or LRF values to create total values. Nevertheless, it is apparent that for these two safety goal surrogate measures, the total risk for all PRA modes would not increase by more than 2 orders of magnitude.

GEH affirms that the individual risk and societal risk goals are maintained with sufficient margin as shown in the preceding table. These results, together with supporting sensitivity studies, lead to the risk insight that the public health and safety is well protected in the ESBWR design, as shown by the PRA analysis.

The staff finds the GEH public health and safety maintenance assertions in the ESBWR PRA to be sound. The staff agrees that the PRA risk and consequence results are within the Commission's safety goals for individual risk, societal risk, and radiation dose, as well as the Commission's CPG. These staff conclusion are conditional on the surrogate risk levels from external events and low-power/ shutdown operations being specifically confirmed by direct release consequences.

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.1.4.3.1.2 Insights

Insights from the reported ESBWR Level 3 PRA results are summarized below.

- The estimated total risk to the public for the ESBWR design is low and acceptable. Offsite risk is very low compared to that of the current generation of operating plants because of a combination of (1) a very low estimated CDF, (2) a low CCFP, and (3) a relatively benign source term associated with the frequency-dominant release category.

- The risk results demonstrate that the ESBWR, for accidents arising from internal events during full-power operation, meets the established consequence-related goals with substantial margin.

- The results for the ESBWR do not explicitly include the contribution to risk from external events. The surrogate risk results for externally initiated events and shutdown operations give confidence that the ESBWR would still meet the Commission's safety goal policy with margin when these additional contributors are included.

- The release category associated with normal containment leakage levels is a low but not negligible contributor to the public risk. It is assigned to every core damage accident.

- The containment failure accident release categories contributing most to the public risk (EVE, BOC, and BYP) have conditional probabilities of occurrence of 0.05 or less. For EVE, this results primarily from the design-driven low probability of high levels of water being present in the LDW just before vessel failure; for BOC, designing to the extent practical all components connected to the RCPB to an ultimate rupture strength at least equal to the full RCPB pressure; and for BYP, the minimization of the number of penetrations.

- The other containment failure accident release categories contributing to the public risk have conditional probabilities of occurrence of 0.01 or less. These low probabilities are largely attributable to the presence of the BiMAC device.

- The applicant has conservatively chosen to designate all containment failures as large release. This is acceptable. The class of severe accidents in which the only release from the containment is at the design leakage rate level is not included in this designation, but it is included in the risk calculations.

The applicant has identified risk insights adequately.

19.1.4.3.2 Significant Accident Sequences and Accident Classes/Release Categories Contributing to Offsite Consequences

Each of the 10 non-zero frequency release categories is represented by one or two severe accident sequences selected and modeled to represent the group of potential severe accidents associated with that release category. The most significant releases from failed containment stem from external steam explosion, BOC, and bypass accident sequences, represented by the release categories EVE, BOC, and BYP.

19.1.4.3.3 Leading Contributors to Risk from the Level 3 Internal Events Probabilistic Risk Assessment

The leading risk contributors listed in this subsection contribute to the risk of the population within 16 kilometers (10 miles) from each of the release categories at 72 hours after the onset of core damage as calculated in the ESBWR Level 3 PRA for internal events at full power. The 72-hour values bound the reported 24-hour values but are not significantly greater. For example, the societal (latent fatality) risk is 1.1×10^{-11}/yr at 72 hours, compared with 9×10^{-12}/yr at 24 hours. The leading risk contributors are the following:

- The whole-body dose at 805 meters (0.5 miles) over the entire dose spectrum from 0.2 Sv to greater than 100 Sv (20 rem to greater than 10,000 rem) is well below the goal of 1×10^{-6}/yr exceedance frequency.

- The containment does not fail following 92 percent of the core damage accidents (TSL release category). TSL releases associated with these non-continuous flow (CF) sequences are estimated to result in about 8 percent of the societal risk within 16 kilometers (10 miles). There is no individual risk contribution from the TSL releases.

- The most significant releases from failed containment stem from external steam explosion, BOC, and bypass accident sequences. The associated risk categories are EVE, BOC, and BYP, which account for 72 percent, 17 percent, and 6 percent of the individual risk, and 50 percent, 18 percent, and 22 percent of the societal risk, respectively.

- The release categories TSL, EVE, BOC, and BYP account for 98 percent of the CDF, 95 percent of the individual risk, and 98 percent of the societal risk.

The applicant has identified leading contributors to risk adequately.

19.1.4.3.4 Risk-Significant Equipment/Functions/Design Features, Phenomena/Challenges, and Human Actions

GEH does not identify any risk-significant equipment, functions, design features, phenomena, challenges, and human actions, as part of the Level 3 ESBWR PRA.. This is acceptable.

19.1.4.3.5 Insights from Uncertainty, Importance, and Sensitivity Analyses

GEH does not report any results for uncertainty or importance analyses for the Level 3 PRA.

The PRA presents sensitivity analyses of the offsite consequences results (considering bounding variations in meteorological conditions, release elevation, release energy (heat and buoyancy), and mission time).

The analysis considers two meteorological conditions. The first, used for the ESBWR Level 3 base case study, is comparable to the ALWR URD meteorological reference data. The second represents a narrower distribution condition. The narrower distribution can represent conservative radiological consequences in certain wind sectors and with certain stability classes.

The analysis studies elevated release with and without buoyant plume energy rise, along with sensitivity on population density. It uses mission times of 24 hours and 72 hours. The results indicate that the variation of certain MACCS2 input parameters, such as the meteorological conditions, would result in minute changes in relation to the measures of the three risk goals. The population dose at 50 miles does not vary much for ground versus elevated release for 24-hour and 72-hour mission time. The risk insights obtained via ground release modeling at 50 miles do not change even with elevated release modeling.

The sensitivity study shows that the three NRC risk goals and the three GEH design risk goals envelop the results of the selected variations of MACCS2 input parameters and assumptions with a margin of several orders of magnitude.

19.1.5 Safety Insights from the External Events Probabilistic Risk Assessment for Operations at Power

The ESBWR PRA analyzed four external event categories, including seismic, internal fires, high winds, and internal floods The methods used in the ESBWR PRA to evaluate external events are acceptable to the NRC because they provide the insights necessary to determine if any design or procedural vulnerabilities exist for these external events. In addition, these methods provide insights needed for design certification requirements, such as ITAAC.

In SECY-93-087, the NRC identified the need for a site specific probabilistic safety analysis and analysis of external events.

19.1.5.1 Results and Insights from the Seismic Risk Assessment

19.1.5.1.1 Methodology and Approach

19.1.5.1.1.1 Summary of Technical Information

The seismic risk assessment uses the PRA-based SMA method to calculate seismic capacities, i.e., high confidence low probability of failure (HCLPF), for important accident sequences and accident classes. The PRA-based seismic margins approach used in this analysis evaluates the capability of the plant to withstand an earthquake of 1.67 times the safe shutdown earthquake (1.67*SSE). The analysis involves the following two major steps: (1) Seismic fragilities; (2) Accident sequence HCLPF analysis. The seismic fragilities of the ESBWR systems, structures, and components are based on generic industry information and ESBWR specific seismic capacity calculations for certain structures. The MIN-MAX method is used in the determination of functional and accident sequence fragilities. Per the MIN-MAX method, the overall fragility of a group of inputs combined using OR logic (i.e., seismic event tree nodal fault tree) is determined by the lowest (minimum) HCLPF input. Conversely, per the MIN-MAX method, the overall fragility of a group of inputs combined using AND logic (i.e., seismic event tree sequence) is determined by the highest (maximum) HCLPF input.

The ESBWR is designed to withstand a 0.5-g SSE. However, it is expected that a plant built to withstand the SSE will actually be able to withstand an earthquake of a larger magnitude. This is because the analyses used for designing the capability of SSCs to withstand the SSE have significant margin. A PRA-based margins analysis systematically evaluates the ability of the designed plant to withstand earthquakes without resulting in core damage. It does not include an estimate of the CDF from seismic events. The margins analysis is a method for estimating the “margin” above the SSE (i.e., how much larger than the SSE an earthquake must be before the safety of the plant becomes compromised).

The capability of a particular SSC to withstand beyond-design-basis earthquakes is measured in terms of the value of the peak ground acceleration (g-level) at which there is a high confidence that the particular SSC will have a low probability of failure (HCLPF). The HCLPF capacity of a certain SSC corresponds to the earthquake level at which, with high confidence (95 percent), it is unlikely (probability less than 5×10^{-2}) that failure of the SSC will occur. An HCLPF value for the entire plant is determined by finding the lowest sequence HCLPF that leads to core damage. It is a measure of the capability of the plant to withstand beyond-design-basis earthquakes without sustaining core damage. The plant HCLPF value, which is assessed from the SSC HCLPF values, has units of acceleration. The risk-based SMA takes no credit for the non-safety-related defense-in-depth systems. Because such systems are not seismic Category I, the analysis conservatively assumes that they become unavailable as a consequence of the seismic initiating event. Because the non-safety-related diesel generators are assumed to be unavailable, and the failure with the lowest HCLPF value that would initiate an accident is the loss of offsite power, the SMA treats all accident sequences as SBO sequences. The analysis investigated and accounted for potential adverse interactions between assumed seismically damaged non-safety-related SSCs and safety-related systems. The event and fault trees developed for the internal events PRA were modified to accommodate seismic events. In this way, the seismic analysis captures the random failures and human errors modeled in the internal events portion of the PRA.

19.1.5.1.1.2 Regulatory Criteria

The NRC has indicated in SECY-93-087 and the associated SRM that a plant designed to withstand a 0.5-g SSE should have a plant HCLPF capacity of at least 1.67 times the acceleration of the SSE (i.e., 0.84 g).

19.1.5.1.1.3 Staff Evaluation

The methodology used to perform the SMA follows an approach the staff has previously accepted for the Individual Plant Examination of External Events (IPEEE) Program and previous design certifications (i.e., the AP1000) and is therefore acceptable.

The PRA-based SMA shows that the ESBWR design can meet the 0.84-g HCLPF value expectation if the seismic capacities of safety system components are qualified to be above the specified acceptable design value of .84 g. In Section 19.2.6 of the DCD, the applicant stated the following:

> The COL holder referencing the ESBWR certified design shall compare the as-built SSC HCLPFs to those assumed in the ESBWR seismic margin analysis shown in Table 19.2-4 [of the DCD]. Deviations from the HCLPF values or other assumptions in the seismic margins evaluation shall be analyzed to determine if any new vulnerabilities have been introduced.

This COL action item is acceptable.

19.1.5.1.2 Significant Accident Sequences and Leading Contributors

19.1.5.1.2.1 Summary of Technical Information

In the systems analysis portion of the SMA, the applicant described a set of potential accident sequences following a seismically induced rupture of the largest pipe in the reactor coolant system (RCS). The applicant assumed that all ac power is lost at the time of the seismic event and that the ac power is unrecoverable. Consequently, these sequences reflect the impact of success and failure of passive safety systems and safety systems that rely only on direct current (dc) control power. The likelihood of components failing randomly was assumed to be insignificant compared to that for seismic-induced failures, and, therefore, the sequences did not include random events.

19.1.5.1.2.2 Regulatory Criteria

The staff has considered the results and insights from the SMA with respect to the Commission’s objectives for new reactor designs stated in Section 19.1.1 of this document. The following objective is especially relevant to the evaluation of results and insights from the SMA:

(1) Identify risk-informed safety insights based on systematic evaluations of the risk associated with the design such that the applicant can identify and describe (a) the design’s robustness, levels of defense-in-depth, and tolerance of severe accidents initiated by either internal or external events and (b) the risk-significance of specific human errors associated with the design.

19.1.5.1.2.3 Staff Evaluation

The staff used the results of the applicant’s risk-informed SMA to identify dominant accident sequences for seismic events.

The applicant’s SMA shows that sequences involving structural failure of buildings or important structures (e.g., control building, RPV support) have larger seismic capacities than those involving failure of mitigating systems and therefore are considered less important. Of the 12 sequences involving failure of mitigating systems that leads to core damage, all have a seismic capacity of 0.84 g. This is the result of using an assumed value (i.e., 0.84 g) for component fragilities and applying the MIN-MAX method for establishing sequence-level seismic capacity. Sequence 15 of the ESBWR SMA is considered to be the most significant of these 12 sequences. This sequence leads directly to core damage following the initiating event and to seismically induced failure of dc power. This is because many of the other mitigating systems depend on dc power in order to perform their functions such that there are no success paths that are independent of dc power. Results from seismic PRAs performed as part of the IPEEE program showed that seismic failures of dc batteries and electrical distribution equipment (e.g., cable trays) were among the most frequently observed dominant contributors to core damage. The staff also considers sequences 8 and 14 of the ESBWR SMA to be potentially dominant because they lead directly to core damage following seismic failure of the ADS. Depressurization is a critical safety function for mitigation of seismic events because the passive ECCS operates at low pressure.

19.1.5.1.2.4 Conclusions

The applicant has successfully identified important safety insights based on the SMA. The applicant's assessment of significant accident sequences and leading contributors for core damage is acceptable.

19.1.5.1.3 Insights from the Uncertainty, Importance, and Sensitivity Analyses

Neither uncertainty analyses, importance analyses, nor sensitivity analyses are available because the applicant performed an SMA rather than a seismic PRA. The explanation of seismic risk using SMA is an approach acceptable to the staff.

19.1.5.2 Results and Insights from the Internal Fires Risk Analysis

19.1.5.2.1 Methodology and Approach

19.1.5.2.1.1 Summary of Technical Information

A fire probabilistic risk assessment (FPRA) is performed taking into account that the specifics of cable routings, ignition sources, and target locations in each zone of the plant are not known at this stage of the plant design. Because of this limitation, the applicant used a simplified conservative and bounding approach. For example, the FPRA assumes the worst effects of fire on all the equipment and systems located in each group of fire areas; that is, any fire in any fire area will cause the worst damage, and a fire ignition in any fire area continues to grow unchecked into a fully developed fire without credit for fire suppression.

The fire risk analysis uses the same PRA models as the internal events evaluation. The specific fire location determines which of the internal events sequences are applicable. These are modified to consider the effects of specific fires and include the possibility of fire propagation through potentially failed fire barriers. Bounding fire initiating event frequencies are used in the analysis, consistent with the nature of the fire analysis.

The ESBWR internal FPRA is performed according to the guidance in NUREG/CR-6850 (EPRI 1011989), "Fire PRA Methodology for Nuclear Power Facilities," issued September 2005.

For the ESBWR FPRA model development, the following tasks in NUREG/CR-6850 apply:

- Task 1: Plant Boundary and Partitioning
- Task 2: FPRA Component Selection
- Task 3: FPRA Cable Selection
- Task 4: Qualitative Screening

- Task 5: Fire-Induced Risk Model
- Task 6: Fire Ignition Frequencies

19.1.5.2.1.1.1 Fire Probabilistic Risk Assessment Assumptions

The fire risk analysis is performed using conservative assumptions, in part because of the stage of the design. The key general assumptions include the following:

- Fire ignition in any fire area may grow into a fully developed fire.

- The analysis does not take credit for any fire suppression systems. Therefore, the analysis assumes that all fires disable all potentially affected equipment in the area.

- The analysis does not take credit for the distance between fire sources and targets.

- The analysis assumes that all fire-induced equipment damage occurs at t = 0.

- Design requirements have been implemented to prevent spurious actuations induced by a single fire in the reactor building. Fire propagation in the reactor building is assumed to result in the inadvertent opening of relief valves (IORV) initiating event.

Since the insights from the FPRA analysis impact the detailed designs, more specific assumptions about each task are made in the FPRA analysis as a result of that process. Section 12.2 of the PRA report describes the detailed assumptions.

19.1.5.2.1.1.2 Task 1: Plant Boundary and Partitioning

The “Electrical Equipment Separation” design specification for the ESBWR provides the basic criteria for separation, both physical and electrical, of redundant safety equipment. ESBWR separation specifications are based on RG 1.75, and Institute of Electrical and Electronic Engineers (IEEE) Standard 384. In addition, the ESBWR design complies with the more stringent NRC policy statement of SECY-89-013, which requires the capability for safe shutdown assuming that all equipment in any one fire area has been rendered inoperable by fire and that reentry to the fire area for repairs and for operator actions is not possible.

The plant is divided into separate fire areas. The redundant cables and equipment are separated by fire barriers to limit any damage caused by a fire and to provide a means to ensure that there is sufficient capacity to perform safety functions in case of fire.

The ESBWR design has 3-hour fire-rated barriers to ensure the following:

- separation of safety-related systems from potential fires in non-safety-related areas that could affect the ability of the safety-related systems to perform their safety functions

- separation of redundant divisions or trains of safety-related systems so that both are not subject to damage from a single credible fire that could consume everything within the given fire area

- fires within the inert containment during plant operation are not credible

- separation of components within a single safety division that could present a fire hazard to other safety-related components

- separation of redundant remote shutdown panels.

The application of these separation criteria ensures an adequate independence of each safety system division, such that a fire in a single fire area can affect only one safety system division. The ESBWR FPRA uses these criteria to support definitions of the major fire areas. ESBWR non-safety-related systems with the potential to adversely affect safety-related systems are designed with similar separation requirements.

The ESBWR FPRA considers only the mitigation of fires without crediting suppression capabilities. The plant is divided into separate fire areas. Fire barriers separate the redundant cables and equipment to limit any damage caused by a fire to ensure that there is sufficient capacity to perform safety functions following a fire event.

The global plant analysis boundary uses all the fire areas defined in DCD, Revision 4, Chapter 9, which covers all the protected area. All fire areas defined in the fire hazard analysis (FHA) are included in the plant boundary. The FHA fire areas include the reactor building, fuel building, control building, turbine building, electrical building, radwaste building, and yard area.

19.1.5.2.1.1.3 Task 2: Fire Probabilistic Risk Assessment Component Selection Criteria

The equipment/component selections are based on the following criteria:

- equipment whose fire-induced failures will contribute to or otherwise cause an initiating event in the FPRA (including spurious actuations)

- equipment that supports the success of mitigating system functions

- equipment that supports the success of operator actions to achieve and maintain safe shutdown (including spurious actuations)

19.1.5.2.1.1.4 Task 3: Fire Probabilistic Risk Assessment Cable Selection

The cable routing assumes divisional separation and is based on current plant general arrangement drawings. The I&C is based on the preliminary design of panels and remote multiplexing units (RMUs). Because of the current limitation of the design details, no detailed circuits are available for evaluation. However, the ESBWR digital I&C system design is required to prevent spurious actuations.

19.1.5.2.1.1.5 Task 4: Qualitative Screening Criteria

The qualitative screening for a fire area uses the following criteria:

- The area does not contain equipment modeled in the PRA (or its associated circuits) identified in FPRA Tasks 2 and 3.

- Fires in the area will not lead to (1) an automatic trip, or (2) a manual trip as specified in fire procedures or plans, emergency operating procedures, or other plant policies, procedures, and practices, or (3) a mandated controlled shutdown as prescribed by plant TSs because of invoking a limiting condition of operation (LCO).

19.1.5.2.1.1.6 Task 5: Fire-Induced Risk Model

The at power FPRA models are based on the Level 1 and Level 2 internal events PRA models. For each fire scenario, the corresponding initiating event in the internal events PRA model is assigned with the evaluation of all failed components in the affected fire area(s).

The calculation of the fire-induced CDF and LRF for each fire scenario requires the determination of the type of initiating events resulting from the fire damage. The initiating events also include the fire-induced damage to mitigating systems credited in the PRA. Mitigating systems in the PRA include both safety and non-safety equipment.

19.1.5.2.1.1.7 Task 6: Fire Ignition Frequencies

The NUREG/CR-6850 methodology is used to calculate the full-power fire ignition frequencies. The specific steps outlined in NUREG/CR-6850 are followed.

19.1.5.2.1.2 Regulatory Criteria

The staff has considered the results and insights from the internal FPRA with respect to the Commission’s objectives for new reactor designs stated in Section 19.1.1 of this document.

19.1.5.2.1.3 Staff Evaluation

The ESBWR internal FPRA is performed according to the guidance in NUREG/CR-6850. The FPRA method documented in this report reflects state-of-the-art fire risk analysis approaches. Methodological issues raised in past fire risk analyses, including IPEEE fire analyses, have been addressed to the extent allowed by the current state of the art. Therefore, the staff finds the use of this approach to perform internal FPRA acceptable.

GEH described the ESBWR plant layout drawing, fire component mapping, and cable routing information in a topical report, NEDO/NEDE-33386, “ESBWR Plant Flood Zone Definition Drawings and Other PRA Support Information,” Revision 0, issued September 2007. DCD Section 9A (Figures 9A.2-1 through 9A.2-47) includes the plant layout drawings for fire areas and fire boundaries. Tables 9A.5-1 through 9A.5-7 in DCD Section 9A list additional information for these fire areas. NEDE/NEDO-33386, Section 4, includes the list of equipment located in each fire area and the cable routing information.

The mapping from fire areas to rooms, then to components and basic events, is based on the current detailed design drawings, which are subject to change. However, the separation criteria are implemented, and this is not expected to change in future modifications to the detailed designs. The cable routing is assumed for the PRA fire model under the guidelines for separation criteria. Although the final cable routing could be different from that assumed in the PRA model, the assumed cable routing in the PRA model is conservative. The staff finds this approach to be acceptable.

The staff also asked specific questions about the locations of the RWCU pumps and trains, requested a list of screened-out fire areas, and requested an explanation of why fires in the yard area and remote shutdown panels were not addressed. The components of RWCU trains are in separate fire areas as shown in Figures 9A.2-1 and 9A.2-10. Table 12.6-2 of ESBWR PRA Section 12A contains a list of screened-out areas. The remote shutdown panels will be located in separate fire areas in the reactor building. Since the FPRA does not take credit for the remote shutdown panels for reasons of conservatism, their location is not critical to the current PRA model. A fire in the switchyard could result in a plant trip if it results in an LOPP. The FPRA model includes such a scenario with a conservative assumption that any fire in the switchyard would result in a reactor trip. The staff finds these responses acceptable.

The staff asked GEH to search for potential smoke propagation paths, identify design and operation features to minimize smoke propagation, and assess the associated risk of smoke propagation.

GEH described the potential smoke propagation in various buildings based on the simplified plant diagram for the ESBWR. Design and operational features used to mitigate the potential risk associated with smoke propagation include following the National Fire Protection Association smoke control guidelines and removing smoke with HVAC systems. GEH described balanced HVAC and safety-related digital control and instrumentation system (Q-DCIS) system designs to address both heat dissipation and smoke removal issues.

GEH is preparing a balanced detailed HVAC system design (i.e., implementing separation criteria of reactor building HVAC subsystems, or coating some of the Q-DCIS circuit boards, or using other equivalent methods to protect them from the postulated smoke damage). According to Appendix T, "Smoke Damage," to NUREG/CR-6850, circuit bridging is the only mode of component failure found to be of potential risk significance. Coating some of the Q-DCIS circuit boards or protecting them by other equivalent methods could significantly reduce potential smoke damage. On the other hand, a detailed HVAC design could implement separation criteria for different fire areas with safety-related equipment, which would result in negligible risks associated with smoke damage even without crediting coating of the Q-DCIS circuit boards. In summary, the risk associated with postulated smoke propagation is negligible if balanced HVAC and Q-DCIS system designs are implemented to address smoke removal issues.

The ESBWR FPRA has evaluated potential fire-induced spurious valve actuations causing LOCA or incorrect valve lineup. A single fire in any fire area will not cause spurious actuation of DPVs, SRVs, or GDCS squib valves to result in a LOCA. The ESBWR I&C system is digital. A spurious signal cannot be induced by the fire damages in a fiber optic cable. The hard wires are minimized to limit the consequences of a postulated fire. Furthermore, two or three load drivers must be actuated simultaneously to actuate the component. To eliminate spurious actuations, these multiple load drivers are located in different fire areas. Therefore, a fire in a single fire area cannot cause spurious actuation.

The ESBWR FPRA has addressed potential fire-induced spurious valve actuations causing ISLOCA. However, the FPRA has not screened and considered two interfacing LOCA systems. The two systems with penetration lines are the main steamline drains upstream of the MSIVs and the feedwater system. Multiple containment isolation valves and drains are configured in different fire areas for the main steamline drain. It is unlikely that a fire could propagate across multiple fire areas and cause spurious actuations on both the containment isolation valves and the downstream valve. For the high/low-pressure interfaces on the feedwater system line A, multiple check valves are included, which prevent the opening of the path even if a spurious actuation should occur after a fire. Moreover, the detailed design has added the monitoring and alarm functions on the line between the check valve and the normally closed isolation valves to check for potential leakage which would indicate valve failure upstream. Therefore, the spurious actuation resulting from a postulated fire has a negligible impact on the ISLOCA evaluations.

New fire propagation scenarios for full-power operation have been postulated based on the plant general arrangement drawings. The FPRA model includes the possibility of fire propagation through potentially failed fire barriers. The failure probabilities of fire barriers are taken from Table 11-3 of NUREG/CR-6850, Volume 2. Fire doors may be open to perform online maintenance. However, Revision 2 of the ESBWR FPRA model does not show this. When the plant is in operation, its risk management program will control the risk increases associated with the open fire doors. This program is required by 10 CFR 50.65(a)(4). Therefore this approach is acceptable to the staff.

Since the main control room (MCR) communicates with the digital control instrumentation system (DCIS) rooms via fiber optic cables, no spurious actuation will originate from an MCR fire. The remote shutdown panels give the operators redundant locations to perform functions related to safe shutdown. However, these actions are for defense in depth. The ESBWR FPRA model for a postulated fire in the MCR does not credit the performance of the compensatory manual actions for safe shutdown. Instead, all operator actions are assumed failed for an MCR fire. The staff finds this acceptable.

The ESBWR FPRA is a bounding analysis that incorporates several conservative assumptions. The fire analysis does not account for the amount of combustible material present or for the distance between fire sources and targets. The analysis assumes that a fire ignition in any fire area grows into a fully developed fire. Therefore, fires are conservatively assumed to propagate unsuppressed in each fire area and to damage all functions in the fire area. Bounding fire initiating event frequencies are used, consistent with the nature of the fire analysis. The staff finds this acceptable.

The ESBWR internal FPRA is performed according to the guidance in NUREG/CR-6850. The FPRA method documented in this report reflects state-of-the-art fire risk analysis approaches and is therefore acceptable. The FPRA model is to be maintained and updated to reasonably reflect the as-built and as-operated plant according to the PRA maintenance program described in Section 19.4 of the ESBWR DCD. The staff documented its review of the applicant's PRA maintenance and update program in Section 19.1.2.3.4.

The ESBWR PRA does not describe the yard and service water structure/building fire layout areas since these areas are site specific. The FPRA uses conservative assumptions to analyze the fire consequences. The COL applicant will supply the fire layout areas for the yard and service water structure/building. Furthermore, the COL applicant will ensure that results of the COL fire analysis are bounded by the ESBWR DC; otherwise, the COL will perform a modified PRA fire analysis. This is acceptable to the staff.

19.1.5.2.1.4 Conclusions

Based on the preceding discussion, the staff concludes that the methodology and results of the internal FPRA described in the ESBWR PRA are acceptable and meet the Commission's goals of less than 1.0×10^{-4}/yr for CDF and less than 1×10^{-6}/yr for LRF.

19.1.5.2.2 Significant Accident Sequences and Leading Contributors

19.1.5.2.2.1 Staff Evaluation

The total CDF for fire events at full power is 8.06×10^{-9}/yr. The total LRF for fire events at full power is 4.83×10^{-10}/yr.

The staff requested that the applicant provide a characterization of the dominant accident sequences and associated major contributors to CDF for each sequence. The following 10 fire scenarios are the leading contributors to core damage. The 10 combined fire scenarios contribute to about 70 percent of the total fire CDF.

(1) A postulated fire in F1311 (Division I Electrical Room) fails Division I safety-related RMUs and load drivers, Division I uninterruptable power supply (UPS) buses, SLCS train A. It also fails Division I safety-related control signals and some DPS control signals.

(2) A postulated fire in F1321 (Division II Electrical Room) fails Division II safety-related RMUs and load drivers, Division II UPS buses, SLCS train B. It also fails Division II safety-related control signals and some DPS control signals.

(3) A postulated fire in F4100 (Turbine General Area) fails condensate and feedwater system, turbine building closed cooling water system (TCCWS), instrument air system, and service air system, and UPS buses in turbine building.

(4) A postulated fire in the switchyard fails offsite power sources, and no recovery of offsite power is assumed.

(5) A postulated fire in F7300 (Service Water Building) fails the plant service water (PSW) system which results in loss of DHR.

(6) A postulated fire in F3301 (Non-1E Electrical Room) fails RWCU train A, FAPCS train A, CRD pump A, condensate and feedwater system, reactor closed cooling water system (RCCWS) train A, potable sanitary water system (PSWS) train A, fire protection system pump U43-P1B, and DPS room.

(7) A postulated fire in F5550 (Upper Electrical Room) fails all train A 6.9-kilovolt switchgear, load centers, and motor control centers (MCCs). It also fails the non-safety train A dc buses and UPS. As a result of the failure of dc buses, diesel generator A is also failed.

(8) A postulated fire in F3110 (Division I DCIS Electrical Room) fails Division I safety-related control signals.

(9) A postulated fire in F3120 (Division II DCIS Electrical Room) fails Division II safety-related control signals.

(10) A postulated fire in F4103 (Reactor Feedwater Pump Room) fails the feedwater system.

The most important fire sequences involve loss of one electrical division as the result of a fire in the reactor building, with common-cause software failures on the digital control systems, and failure of GDCS injection because of CCF of check valves, and operator failure to recognize the need to provide low-pressure injection.

The quantification of the LRFs is similar to the CDF calculations with the addition of the Level 2 fault tree models and phenomenological point estimates. The fire-induced risk model used for Level 1 quantification is not changed since the component selection and cable selection tasks have already considered all components, including the Level 2 components.

The leading contributors to the LRF are similar to those for the CDF except that a postulated fire in Cable Tunnels A and B (F9150 and F9160) contributes to approximately 35 percent of the total LRF.

Based on the preceding discussion, the staff concludes that the applicant has provided an adequate discussion of dominant accident sequences.

19.1.5.2.3 Risk-Significant Functions/Features, Phenomena/Challenges, and Human Actions

19.1.5.2.3.1 Staff Evaluation

The ESBWR design features safety system redundancy and physical separation by fire barriers. The design ensures that in all cases a single fire limits damage to a single safety system division or defense-in-depth system. Fire propagation to neighboring areas presents a relatively minor risk contribution.

The ESBWR internal events PRA model assumes that both trains of the SLCS are required to mitigate the accident consequences from the ATWS sequences. Consequently, a fire that affects a single train of the SLCS is considered risk important.

Fire in the control room traditionally requires the operator to take actions to control the plant manually. One feature relevant to the ESBWR design is that a fire in the control room does not affect the automatic actuations of the safety systems. Additionally, the existence of remote shutdown panels allows the opportunity to perform manual actuations for failed automatic actuations that may occur.

Similar to the internal events analysis, the FV importance values for fires are low, which indicates a balanced risk profile. The most important component failures are the ICS vent valves failing to open.

Besides the ability of the BiMAC device to cool the molten core, there are no insights from SSCs or operator actions in large-release fire sequences.

The applicant has successfully identified risk significant functions and features.

19.1.5.2.4 Insights from the Uncertainty, Importance, and Sensitivity Analyses

19.1.5.2.4.1 Staff Evaluation

Sensitivity analysis was performed for the Level 1 fire using focused PRA studies of (1) failing all non-safety systems and (2) failing all non-safety systems except those designated as RTNSS. The Level 1 focus fire with all non-safety systems failed generated a CDF of $1.15x10^{4}$/yr; the RTNSS generated a CDF of $2.40x10^{-7}$/yr. The results for the focus fire sensitivity showed significant impact to the CDF with the failure of non-safety systems both with and without RTNSS. The inclusion of the RTNSS in the model reduces the CDF by approximately 2 orders of magnitude compared to crediting safety-related systems only. Based on the Level 1 fire focus sensitivities CDF results, the NRC goal of $1x10^{-4}$/yr CDF is met for both the baseline Level 1 fire model and the RTNSS sensitivities. The focus fire case CDF with all non-safety systems failed does not meet the NRC goal. However, the fire analysis is very conservative with no credit for fire suppression or fire severity factors.

The Level 2 focus fire with all non-safety systems failed generated an nTSL (non-technical-specification leakage, which is equivalent to LRF) release frequency of $1.15x10^{-4}$/yr and a CDF of $1.15x10^{-4}$/yr. The RTNSS generated an nTSL release frequency of $4.72x10^{-8}$/yr and a CDF of $2.40x10^{-7}$/yr. The results for the focus sensitivity showed significant impact to nTSL release frequency with the failure of non-safety systems both with and without RTNSS. The results showed a decrease of 4 orders of magnitude in the nTSL frequency with the RTNSS available compared to safety-related systems only. Based on the Level 2 fire focus sensitivities nTSL results, the NRC goal of $1x10^{-6}$/yr LRF is met for RTNSS, but this goal is not met for the focus Level 2 fire with all non-safety systems failed.

Tables in Chapter 11 of the PRA report present the results of FPRA sensitivity studies in the column entitled "Difference." These tables include 11.3-4, 11.3-6, 11.3-8, 11.3-11, 11.3-19, 11.3-20, 11.3-22, 11.3-23, 11.3-24, 11.3-25, 11.3-28, 11.3-30, 11.3-32, 11.3-34, 11.3-36, 11.3-37, 11.3-38, and 11.3-39. Section 11 does not define "Difference." The staff cannot reproduce some of the results and is concerned that there may be some errors in the calculation of "Difference." The staff has issued RAI 19.1-160 to ask GEH for resolution. **RAI 19.1-160 is being tracked as an open item**.

In addition to the focused PRA studies, a series of sensitivities were conducted to determine the impact to CDF and LRF in the full-power and shutdown FPRA models from the uncertainties in the model assumptions. The full-power fire model sensitivity studies are grouped as follows:

- plant partitioning
- fire risk in transition modes
- fire ignition frequencies
- separation criteria
- fire barrier failure probabilities

The results of the plant partitioning sensitivity study indicated that DPS is critical in mitigating the fire risks, which warrants the separation of the DPS cabinet(s) from other cabinets in Room 3301. The risk increases associated with the merging of Rooms 3301 and 3140 into a single fire area are moderate. In both cases, the resulting total fire risks are still at least 3 orders of magnitude lower than the NRC goals for CDF and LRF (i.e., $1x10^{-4}$/yr for CDF and $1x10^{-6}$/yr for LRF).

The sensitivity study of fire in transition modes indicated that fire area F1170 (drywell and containment fire area) warranted further study. This room is inert during full-power operation (Mode 1) and de-inerted in Mode 2, 3, or 4. The results of the sensitivity studies indicate that total baseline CDF and LRF in these modes are at least 3 orders of magnitude below the goals.

The results of the fire ignition frequencies sensitivity study confirmed that fire ignition frequencies used in the baseline FPRA model are conservative. The staff finds this acceptable.

The results of the separation criteria sensitivity analysis showed the importance of the RTNSS requirements for RCCWS and PSWS to ensure separation criteria.

The results of the fire barrier failure sensitivity/importance study indicated that the risk increases with several fire barrier failures are significant. The three most risk-significant increases for barrier failures are the barrier between Cable Tunnels A and B, the barrier between the Non-1E Electrical Room and DPS room, and the barrier between the Division IV Electrical Room and the area housing RWCU train B, CRD Panel D, part of DPS panels generating control signals.

The results of importance measures for the at-power fire CDF were generated. The results confirmed the importance of components in cutsets of the top fire sequences.

By crediting the DPS and ARI functions along with the safety-related systems, the ESBWR LRF can be significantly reduced to satisfy the safety goal of 1×10^{-6}/yr for LRF in the Level 2 fire model.

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.1.5.3 <u>Results and Insights from Internal Flooding Analysis</u>

19.1.5.3.1 Methodology and Approach

19.1.5.3.1.1 Summary of Technical Information

The objective of the ESBWR internal probabilistic flood analysis is to identify and provide a quantitative assessment of the CDF and releases that result from internal flooding events. The floods may be caused by large leaks resulting from the rupture or cracking of pipes, piping components, or water containers such as storage tanks. Another possible flooding cause is the operation of fire protection equipment.

A flooding event may result in an initiating event and may also disable mitigating systems. Thus, buildings containing mitigating equipment credited in the PRA accident sequence analysis, or equipment whose loss could cause an initiating event, are of interest in the flooding analysis.

The ESBWR analysis considers flood scenarios in the following buildings:

- reactor building
- control building
- fuel building
- turbine building
- electrical building
- service water building
- circulating water pump-house
- fire protection enclosure
- tunnels and galleries connected with the buildings listed above

The study does not consider floods in the remaining ESBWR buildings because the flood water cannot propagate to any of the above buildings.

Buildings are divided into flooding zones and are further subdivided into systems that have the potential to cause flooding within the flooding zone. The analysis does not consider flood zones that do not contain flood sources and do not have floods propagating to the zone. Flood zones that do not cause a reactor trip at power or do not contain mitigating equipment modeled in the PRA are also screened from further analysis. Finally, if the flood zone contains mitigating equipment, such as sump pumps, that would prevent unacceptable flood levels, the flood zone is not analyzed further.

Section 13.2 of the ESBWR PRA lists the assumptions used in the flooding analysis. The major assumptions are the following:

- Nonqualified submerging equipment (motors or solenoids for valves, control cabinets, and circuitry) is assumed to result in equipment failure.

- Motor-operated valves (MOVs) require the application of current to the motor to change the valve position. Without power, the valve will remain in its current position. Flooding and/or spraying of an MOV will therefore cause the valve to fail as is.

- Passive components, such as check valves, pipes, and tanks, are not considered to be vulnerable to flooding effects.

- Flooding has no effect on CCFs.

- Water in a stairwell or propagating into a stairwell preferentially continues to travel down the stairwell as opposed to propagating under a door leading outside the stairwell.

- The mission time of the active equipment credited in the flooding risk analysis is 24 hours. This is the same time used in the internal events PRA.

- The flooding analysis does not consider concurrent flooding events from different sources.

- Components that are environmentally qualified inside containment are considered to be invulnerable to the effects of flooding because they are qualified for a post-LOCA environment inside containment. Environmentally qualified equipment outside containment may not be qualified to a severe environment.

- The internal flooding analysis uses the same system success criteria as used in the internal events PRA.

- Electrical connections in the termination boxes on the containment wall are adequately protected to prevent flood-induced failure.

- Fire doors are not watertight.

- Walls are assumed to be capable of withstanding the expected maximum flood loading. Therefore, walls are assumed to remain intact throughout a flooding event.

- It is assumed that electrical circuit fault protection has been designed to defend plant electric circuits via protective relaying, circuit breakers, and fuses. Therefore, loss of a component because of flooding will not result in the loss of the bus that supplies power to the affected component.

- For floor drains, appropriate precautions such as check valves, backflow prevention, and siphon breaks are assumed to prevent backflow and any potential flooding.

- It is assumed that the doors connecting the control and reactor buildings with the electrical building galleries are watertight; that flooding of the galleries up to the ground-level doors would generate an alarm in the control room, and that procedures direct the immediate closure of the doors upon receipt of an alarm.

- It is assumed that the operation of the components located in containment would not be affected in a LOCA or if the drywell is flooded to a level equivalent to the level of the suppression pool.

- Equipment located in the yard is not considered susceptible to internal flooding damage.

A screening was performed based on a general review of all systems for the ESBWR. This screening removed systems that would not be considered flood sources from further consideration. After screening, the following plant systems are considered as potential flood sources at power:

- NBS
- CRDS
- SLCS
- FAPCS
- RWCU/shutdown cooling (SDC) system
- resin transfer system
- turbine main steam system
- condensate and feedwater system
- heater drain and vent system
- condensate purification system
- moisture separator reheater system
- extraction steam system
- circulating water system
- makeup water system
- condensate storage and transfer system
- PSW system
- FPS
- station water system
- auxiliary boiler oil storage and transfer system

Systems inside containment considered in the flooding analysis as potential flood sources are those in which a break would cause a LOCA. Because LOCA scenarios in containment are already modeled in the internal events PRA analysis, they are not analyzed in the internal flooding analysis. Therefore, no flood scenarios in containment are analyzed further in the at-power internal flooding analysis.

The initiating event frequency was calculated for each flood zone by summing the frequencies for flood components and piping for the system under consideration. At-power flooding frequencies are included if the failure of the system directly causes a reactor trip, or the flooding caused by the failure fails equipment which leads to a reactor trip, or if PRA-related equipment would likely be affected.

For postulated flood events occurring at power, the general transient initiating event category and associated accident sequence logic are used to model the accident sequence progression. The calculated flood initiator frequency and associated equipment impacts are propagated through the general transient Level 1 internal events accident sequence logic for the flood scenario. A Level 2 analysis was also performed for the flooding scenarios.

19.1.5.3.1.2 Regulatory Criteria

The staff has considered the results and insights from the internal flooding PRA with respect to the Commission's objectives for new reactor designs stated above in Section 19.1.1.

19.1.5.3.1.3 Staff Evaluation

GEH has performed the PRA flooding analysis. The calculated flood initiator frequency and associated equipment impacts are propagated through the general transient Level 1 internal events accident sequence logic for the flood scenario. NEDE/NEDO-33386, "ESBWR Plant Flood Zone Definition Drawings and Other PRA Support Information," Revision 0, provides a list of the equipment located in each flooding area that is credited in the PRA for accident mitigation. The equipment includes safety as well as non-safety components.

The ESBWR PRA provides a list of screened flooding areas. The analysis explains why certain areas are screened-out.

NEDO/NEDE-33386 lists all unscreened flooding sources located in an unscreened area. Flooding initiating event frequency in the flooding zone is based on all potential sources including pipes, pumps, valves, tanks, heat exchangers, and expansion joints within the flooding zone.

Components that are environmentally qualified inside containment are considered invulnerable to the effects of flooding because they are qualified for a post-LOCA environment inside containment.

Flooding propagates between areas. Where propagation is likely, it has been included in the model unless adequate water removal is available (i.e., via sump pumps) to prevent flooding of the target area. Systems that do not have enough capacity to flood an area have been removed from consideration. The analysis considers aspects that affect flood progression in each building. Depending on the building and the origin of the flood, the analysis considers the following aspects that affect flood progression:

- automatic flood detection systems
- automatic systems to terminate flooding
- watertight doors to prevent the progression of flooding

- sump pumps

- other design or construction characteristics that contribute to minimizing the consequences of flooding

The NEDO/NEDE-33386 flooding mapping report considers the scenario from main steam and feedwater pipes located in the steam tunnel propagating to the reactor building.

The mission time of the active equipment credited in the flooding risk analysis is 24 hours. This is the same as the time used in the internal events PRA.

Breaks in support systems like service water, RCCW, and turbine building secondary closed cooling water (TCCW) have been revised and considered in the internal flooding analysis instead of being assigned the same consequences as the failure of the systems themselves as described in the previous revision of the applicant's PRA.

A recovery factor of 0.01 was applied to the circulating water flooding scenario in the turbine building to account for automatic closure of isolation valves and automatic trip of circulating water pumps.

The internal probabilistic flood analysis takes into account equipment locations based on existing plant layout drawings. It assumes that the pipe routed to or from the equipment would follow certain logical paths. For example, pipe is routed through pipe chases in battery rooms instead of being routed through the battery room. Another logical path would be to take the shortest route, which reduces piping and fabrication cost.

The internal flooding PRA model is to be maintained and updated to reasonably reflect the as-built and as-operated plant according to the PRA maintenance program described in Section 19.4 of the ESBWR DCD. The staff's review of the applicant's PRA maintenance and update program appears in Section 19.1.2.3.4 of this document.

NEDE/NEDO-33386 does not describe the yard and service water structure/building flooding areas since these areas are site specific. The internal flooding PRA uses conservative assumptions to analyze the flooding consequences

19.1.5.3.1.4 Conclusions

Based on the preceding discussion, the staff concludes that the methodology and results of the internal flooding risk analysis described in the ESBWR PRA are acceptable and meet the Commission's goals of less than 1.0×10^{-4}/yr for CDF and less than 1×10^{-6}/yr for LRF.

19.1.5.3.2 Significant Accident Sequences and Leading Contributors

19.1.5.3.2.1 Staff Evaluation

The total CDF for full-power internal flooding events is 1.62×10^{-9}/yr. The total release frequency for internal flooding events excluding TSL at full power is 2.7×10^{-10}/yr.

The following 10 flooding scenarios are the leading contributors to core damage. The 10 combined flooding scenarios contribute to about 18 percent of the total flooding CDF.

(1) Flooding in the reactor building at +17500 mm (57.4 feet) elevation caused by a pipe leak of one standby liquid control (SLC) (i.e., SLCS failed) and CCF of rods to insert result in core damage.

(2) Flooding in the service water building caused by a pipe leak of Train A of the service water system (Train A of service water system failed) and CCF of DCIS software and CCF of DPS processors result in core damage.

(3) Flooding in the service water building caused by a pipe leak of Train B of the service water system (Train B of service water system failed) and CCF of DCIS software and CCF of DPS processors result in core damage.

(4) Flooding in the fuel building storage pool at -11500 mm (-37.7 feet) elevation caused by a pipe leak of one SLC train (SLCS failed) and CCF of rods to insert result in core damage.

(5) Flooding in the reactor building at -6400 mm (-21 feet) elevation caused by pipe leak of the control rod system (control rod drive system failed) and CCF of reactor protection system and CCF of DPS processors result in core damage.

(6) Flooding in the service water building caused by a pipe leak of Train A of the service water system (Train A of service water system failed), noncondensable gas failure to vent in the ICS, CCF of DPS processors and CCF of check valves in the GDCS, and operator failure to recognize need of depressurization result in core damage.

(7) Flooding in the service water building caused by a pipe leak of Train A of the service water system (Train A of service water system failed), noncondensable gas failure to vent in the ICS, spurious DPS processor software failure, CCF of check valves in the GDCS, and operator failure to recognize need of depressurization result in core damage.

(8) Flooding in the service water building caused by a pipe leak of Train B of the service water system (Train B of service water system failed), noncondensable gas failure to vent in the ICS, CCF of DPS processors and CCF of check valves in the GDCS, and operator failure to recognize need of depressurization result in core damage.

(9) Flooding in the service water building caused by a pipe leak of Train B of the service water system (Train B of service water system failed), noncondensable gas failure to vent in the ICS, spurious DPS processor software failure, CCF of check valves in the GDCS, and operator failure to recognize need of depressurization result in core damage.

(10) Flooding in turbine building at -1400 mm (-4.6 feet) elevation caused by a pipe leak of the circulating water system (circulating water system failed), spurious DPS processor software failure, CCF of check valves in the GDCS, and operator failure to recognize need of depressurization result in core damage.

The CET release category frequencies are summarized as follows:

Release Category	Frequency

TSL	$1.41x10^{-9}$/yr
Containment bypass	$1.29x10^{-10}$/yr
Overpressure because of failure of short-term CHR	$6.40x10^{-11}$/yr
Wet CCI	$1.20x10^{-11}$/yr
Overpressure because of failure of long-term CHR	$1.00x10^{-12}$/yr

The combined release frequency excluding TSL is about $2.06x10^{-10}$/yr.

19.1.5.3.3 Risk-Significant Functions/Features, Phenomena/Challenges, and Human Actions

19.1.5.3.3.1 Staff Evaluation

Because of the inherent ESBWR flooding mitigation capability, only a few flooding-specific design features are key in the mitigation of significant flood sources. These features include the following:

- using watertight doors in the accesses to tunnels and galleries from the control and reactor buildings
- not locating flood sources with a significant volume of water in the electrical equipment rooms located in the reactor building
- locating an automatic circulating water system pump trip and valve closure on high water level in the condenser pit

The most important flood sequences during at-power conditions involve service water piping leaks at the service water structure which result in the loss of an entire train. Another important sequence assumes that both SLCS trains are required for success because of uncertainties associated with the SLCS flow model.

During the initial phase of the ESBWR design, a significant flood risk in the control building because of a break in FPS piping was identified. Based on this PRA insight, the design specifications now require that the FPS pipes and fire hose stations be relocated outside of the control building such that a piping failure does not result in a significant flood.

The important flooding sequences do not impose additional challenges to any of the PCCSs or the BiMAC. Therefore, the insights into internal events containment performance can be directly used for internal flood sequences.

The applicant has not determined offsite consequences for flooding events because of the bounding method that is used to calculate the flood CDF and its very low value compared to that of the internal events CDF.

19.1.5.3.4 Insights from the Uncertainty, Importance, and Sensitivity Analyses

19.1.5.3.4.1 Staff Evaluation

Sensitivity analysis was performed for the Level 1 internal flooding using focused PRA studies of (1) failing all non-safety systems and (2) failing all non-safety systems except those designated as RTNSS. The Level 1 focus flood with all non-safety systems failed generated a CDF of 1.15×10^{-5}/yr; the RTNSS generated a CDF of 9.06×10^{-9}/yr. The results for the focus flood sensitivity showed significant impact to the CDF with the failure of non-safety systems both with and without RTNSS. The inclusion of RTNSS in the model reduces the CDF by approximately 3 orders of magnitude compared to the CDF when crediting safety-related systems only. Based on the Level 1 flood focus sensitivities results, both the focus flood model and RTNSS sensitivities meet the NRC goal of 1×10^{-4}/yr CDF.

The Level 2 focus flood with all non-safety systems failed generated an nTSL (equivalent to LRF) release frequency of 4.49×10^{-6}/yr and a CDF of 1.15×10^{-5}/yr. The RTNSS generated an nTSL release frequency of 1.23×10^{-9}/yr and a CDF of 9.06×10^{-9}/yr. The results for the focus sensitivity showed significant impact to both nTSL release frequency and CDF with the failure of non-safety systems both with and without RTNSS. The results showed about 2 orders of magnitude decrease in the nTSL frequency with RTNSS available compared to the frequency when crediting safety-related systems only. Based on the Level 2 flood focus sensitivities nTSL results, the NRC goal of 1×10^{-6}/yr LRF is met for the RTNSS but not for the focus flood model which does not credit non-safety systems.

In Section 11.3.4.3 of the applicant PRA it is stated that "The focus Level 2 flood generated a nTSL release frequency of 4.49E-4/yr…." However, Table 11.3-30 shows a frequency of 4.49×10^{-6}/yr. The focus Level 2 flood frequency is either 4.49×10^{-4}/yr or 4.49×10^{-6}/yr. In either case, the NRC goal is not met for the Level 2 focus flood.

The same section states that the "NRC goal of 1E-06/y4 LRF is met for both focus and RTNSS…." This is not an accurate statement of the result of focus Level 2 internal flooding as discussed in the previous paragraph.

The figure "1E-06/y4" appears to be a typographical error. The NRC goal for LRF is 1×10^{-6}/yr.

The staff issued RAI 19.1-161 to ask GEH to correct this typographical error and revise the quoted sentence to reflect that the goal has not been met. **RAI 19.1-161 is being tracked as an open item**.

The results of importance measures for the at-power internal flooding CDF confirmed that components in cutsets of top flooding sequences are important from a risk perspective.

By crediting the DPS and ARI functions along with the safety-related systems, the ESBWR LRF can be significantly reduced to satisfy the safety goal of 1.0×10^{-6}/yr for LRF in the Level 2 flooding model. By crediting the RTNSS, the safety goal for LRF in the Level 2 flooding analysis can also be met.

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.1.5.4 Results and Insights from High-Winds Analysis

19.1.5.4.1 Summary of Technical Information

The NRC's review of the ESBWR high-winds risk assessment is based on the results reported in Chapter 14 of the ESBWR PRA report. GEH developed the risk assessment of high winds for the ESBWR for tornado and hurricane initiators. Straight winds are of lesser velocity and are assumed to pose minimal challenges to the plant design. The risk assessment and the NRC's evaluation encompass plant operation at power, in cold shutdown, and in refueling modes. Mode 5 begins when the RCS temperature drops to or below 93.3 °C (200 °F) while the plant is shutting down. Mode 6 begins when one or more reactor vessel head closure bolts is less than fully tensioned. The tornado and hurricane risk analyses performed by GEH are similar and are primarily deterministic. The evaluation of plant risk from high winds at power uses the internal events PRA accident sequence structure (i.e., the LOPP event tree), system fault trees, and success criteria for LOPP events. The ESBWR shutdown risk assessment is used similarly to calculate shutdown high-winds risk. The high-winds risk assessment assumes that when the reactor well is flooded (Mode 6 (Flooded)) the risk associated with LOPP is negligible, because of the large amount of water stored above the core. This water is assumed to ensure core cooling over a long period (i.e., significantly greater than 24 hours).

Table 14-2 of the PRA report estimates the CDF from an at-power F2 to F3 tornado strike on an ESBWR to be $4x10^{-13}$/yr. The estimated CDF for F4 to F5 tornadoes at power is $5x10^{-11}$/yr. The estimated CDF from all tornadoes when an ESBWR is shut down is $1x10^{-11}$/yr. The estimated CDF from an at-power hurricane is $1.3x10^{-9}$/yr. The estimated CDF from a hurricane during shutdown modes is $1.2x10^{-9}$/yr (a hurricane-induced loss of offsite power in Mode 5 or Mode 5 (Open) accounts for over 98 percent of the shutdown high-winds CDF).

The ESBWR high-winds CDF accounts for the duration (in hours) of operation in Modes 5 and 6 per outage and the anticipated calendar outage frequency of one refueling outage every 2 years. Therefore, the NRC believes the high-winds CDF can be added to the full-power internal events CDF.

These reported tornado- and hurricane-induced CDF estimates should not be considered the true expected risk from high-winds events. These estimates are so low that they tend to lose their physical meaning and challenge one's degree of belief. However, these numerical estimates do show that if an ESBWR plant is built as described in the DCD, the design should be very robust with respect to tornado and hurricane strikes. At frequencies in the range of $1x10^{-8}$/yr (and even higher), uncertainties associated with areas not modeled or not modeled well in the PRA (e.g., errors of commission, rare events, construction errors, design errors) overshadow the absolute value of estimates. In designing the ESBWR, GEH has reduced or eliminated many of the sequences (such as ATWS or SBO) that are leading contributors to core damage in most operating BWRs. The consequence of these design decisions is that less likely events or events that normally are not as well understood or modeled begin to become more important contributors. For some low-frequency initiators normally evaluated in PRAs by bounding analyses, the bounding analyses themselves can give high enough CDF estimates compared to the low reported internal events CDF estimates that these formerly unimportant contributors to risk become potentially much larger (percentage-wise) contributors to CDF and risk (although low in absolute value). Similarly, some low-frequency initiators may have such a high level of uncertainty (e.g., modeling uncertainty, data uncertainty) as to raise the potential importance of the initiator for a design where the CDF estimates are significantly reduced for sequences normally dominant in PRAs.

GEH concludes from its analysis that the CDF resulting from high winds is not a significant contributor to ESBWR core damage risk.

19.1.5.4.1.1 Methodology and Approach for Tornadoes

GEH based its high-wind risk analysis for tornadoes on the premise that plant structures built to seismic Category I and II requirements are invulnerable to the direct effects of tornado winds and tornado missiles. The assessment assumes that, following a strike by an F2 or greater tornado, the equipment housed in these structures will operate with normal equipment failure rates. The assessment also assumes LOPP to the plant and loss of certain structures and their contents depending on the tornado windspeed. The risk assessment involves the following:

- tornado hazard frequency
- tornado-induced plant effects
- calculation of tornado-induced CDFs and release frequencies

The ESBWR high-winds tornado analysis uses data from the National Oceanic and Atmospheric Administration (NOAA) to estimate the tornado hazard frequency. The risk assessment segregates the raw data from the NOAA Web site into three bins—F0 and F1 tornadoes (by the Fujita Scale[5] or F-scale), F2 and F3 tornadoes, and F4 and F5 tornadoes. The number of F0 and F1 tornadoes was discarded as these tornadoes are assumed to not significantly damage structures on site and to cause only LOPP. The DCD indicates that the frequency of such power losses is captured under the initiating events for LOPP. The F2 and F3 raw numbers (i.e., the number of observed F2 and F3 tornadoes in the observation period) divided by the area of the continental U.S., and divided by the number of years of data (56), are used to estimate the frequency of such an event. Similarly, the risk assessment estimated the frequency of an F4 or F5 tornado. These calculations give estimates of tornado occurrence per square mile in the United States per year. The area associated with an ESBWR site is assumed in the risk assessment to be approximately 0.14 mi^2. By multiplying the frequency of tornado occurrence per square mile by 0.14 mi^2, the risk assessment arrives at an annual tornado strike occurrence frequency for the site. The at-power values used in the risk assessment for a strike on the site are $7x10^{-6}$ occurrences per year for F2/F3 tornadoes and $4x10^{-7}$ occurrences per year for F4/F5 tornadoes.

The frequency of occurrence of tornado strikes was further partitioned in the risk assessment to take into account the amount of time that the reactor will be in various shutdown modes. For Mode 5, the tornado occurrence frequency is the number of hours in Mode 5, times the at-power tornado occurrence frequency, divided by the number of hours in a year, divided by the frequency of refueling. The assumed number of hours in Mode 5 per refueling is 192. It is planned that refueling will occur every 2 years. This results in an occurrence estimate when the plant is in Mode 5 of $8x10^{-8}$/yr for F2/F3 tornadoes and $4x10^{-9}$/yr for F4/F5 tornadoes using the frequencies in the ESBWR risk assessment. For Mode 5 (Open), the tornado occurrence frequency is the number of hours in Mode 5 (Open), times the at-power tornado occurrence frequency, divided by the number of hours in a year, divided by the frequency of refueling. The assumed number of hours in Mode 5 (open) per refueling is 48. This results in an occurrence estimate when the plant is in Mode 5 (open) of $2x10^{-8}$/yr for F2/F3 tornadoes and $1x10^{-9}$/yr for F4/F5 tornadoes in the ESBWR risk assessment. For Mode 6 (Unflooded), the tornado occurrence frequency is the number of hours in Mode 6 (Unflooded), times the at-power tornado occurrence frequency, divided by the number of hours in a year, divided by the frequency of refueling. The assumed number of hours in Mode 6 (Unflooded) per refueling is 59. This results in an occurrence estimate when the plant is in Mode 6 (Unflooded) of $2x10^{-8}$/yr for F2/F3 tornadoes and $1x10^{-9}$/yr for F4/F5 tornadoes in the ESBWR risk assessment.

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[5] Damage caused by a tornado is rated by the Fujita Scale. The higher the Fujita Scale number, the faster the rotational speed and more destructive the tornado.

19.1.5.4.1.2 Methodology and Approach for Hurricanes

As for tornadoes, GEH based its high-wind risk analysis for hurricanes on the premise that plant structures built to seismic Category I and II requirements are invulnerable to the direct effects of hurricane winds. The assessment assumes that the equipment housed by these structures will operate with normal equipment failure rates. The assessment also assumes LOPP to the plant and loss of various structures and their contents depending on the hurricane windspeed. The risk assessment evaluation involves the following:

- hurricane hazard frequency
- hurricane-induced plant effects
- calculation of hurricane-induced CDFs and release frequencies

The ESBWR high-winds hurricane analysis uses data from NUREG/CR-6890, to estimate the hurricane hazard frequency. The ESBWR risk assessment took the number of losses of offsite power that occurred at nuclear power plants as the result of hurricanes during this 19-year period and divided it by the number of reactor critical years that coastal nuclear power plants had operated during this same period. The risk assessment takes this estimate (1.5×10^{-2} per reactor calendar year) as the frequency of a hurricane striking a coastal plant when the plant is at power, causing a loss of offsite power, and potentially causing other damage that might lead to core damage and fission product release. The key assumption in the risk assessment for this evaluation is that hurricanes have no possibility of damaging seismic Category I, seismic Category II, or RTNSS structures.

The frequency of hurricane occurrence (i.e., the frequency of hurricane strikes causing loss of offsite power) was further partitioned to take into account the time the reactor might be in various shutdown modes. For Mode 5, the hurricane occurrence frequency is the assumed number of hours in Mode 5 times the at-power hurricane occurrence frequency, divided by the number of hours in a year, divided by the frequency of refueling. The assumed number of hours in Mode 5 per refueling is 192. It is planned that refueling will occur every 2 years. This results in an occurrence estimate when the plant is in Mode 5 of 1.7×10^{-4} occurrences per year for hurricanes. For Mode 5 (Open), the hurricane occurrence frequency is the number of hours in Mode 5 (Open) times the at-power hurricane occurrence frequency divided by the number of hours in a year, divided by the frequency of refueling. In addition, because of the long advance warning of an approaching hurricane, the risk assessment assumes that the plant will have adequate time to set the reactor head in place, which results in a Mode 5 (Open) configuration. Because of this assumption, the risk assessment has added the shutdown hurricane Mode 6 (Unflooded) high-wind strike frequency to the Mode 5 (Open) frequency. The assumed number of hours in Mode 5 (open) per refueling is 48, and the number of hours in Mode 6 (Unflooded) is 59. This results in an occurrence estimate when the plant is in Mode 5 (Open) of 9.3×10^{-5}/yr.

19.1.5.4.1.3 Risk-Significant Functions and Features

Listed below are key ESBWR design features and functions that significantly reduce the expected CDF associated with hurricane strikes as compared to the CDF for operating BWR designs. The risk-significant functions are primarily the same as those identified for LOPP for internal events, as well as improvements in the hardening of the structures housing the systems and components providing long-term DHR capabilities.

- The ESBWR design calls for a significant amount of water to be stored over the core that is available for gravity-driven core cooling. This is not true for most operating BWRs.

- The exterior walls of the ESBWR reactor building are generally thicker than those of the reactor buildings of operating BWRs.

- The ICs in the ESBWR are wholly contained inside secondary containment, and the exterior walls surrounding the ICs are generally thicker than those walls protecting ICs at older operating BWRs.

- The ESBWR long-term DHR design relies on more robust dc power (i.e., the batteries have longer design battery life) compared to operating reactors where dc power generally will last only 4 to 8 hours, compared to 72 hours for the ESBWR design.

- The ESBWR design depends in great part on gravity injection rather than turbine-driven pumps to provide core cooling.

- The ESBWR design has dedicated refill lines for the ICs unlike older operating BWRs with ICs.

- The ESBWR ICs store a larger water supply per megawatt over core than do older plants with ICs.

- The ESBWR design has eliminated or reduced many contributors to CDF resulting from extended loss of offsite power. This has resulted in the CCF of digital I&C systems becoming an important contributor in hurricane-induced CDF. While the CCF of digital I&C systems is a larger contributor to CDF as a percentage at the ESBWR than at operating plants, the absolute value of the contribution to CDF from this source is similar for operating and ESBWR designs.

19.1.5.4.1.4 Significant Sequences and Leading Contributors

The CCFs of the following SSCs are significant based on the reported RAW values (all near or in excess of 1000) in Tables 14.5 and 14.6 of Revision 2 of the PRA report:

- containment vacuum breakers
- containment vacuum breaker isolation valves
- inverters in the uninterruptible ac power supply
- batteries in the dc power system
- IC heat exchangers
- logic units in the DPS
- DPVs
- GDCS injection valves
- IC condensate return valves
- check valves in the GDCS
- software
- DPS processors
- air-operated scram valve (AOV)-126
- control rods insertion
- DPS load drivers

Of the top 100 cutsets for high-winds events, two were caused by tornado strikes. The rest were caused by hurricane-induced LOPP. The only SSC identified as an important contributor to risk in the two LOPP cutsets associated with tornadoes was the CCF of software.

The following important SSCs and human actions were identified in the top 30 hurricane-induced cutsets as contributing to core damage:

- software CCF
- CCF of check valves in the GDCS
- CCF of DPVs
- operator failure to recognize need for depressurization
- operator failure to recognize need for low-pressure makeup after depressurization
- control rods failure to insert
- failure of any SRV to reclose following ATWS
- misposition of valve F334

The following SSCs have significant CCFs based on the reported RAW values in Tables 14.5 and 14.6 of Revision 2 of the PRA report. The RAW values are all near or in excess of 1000.

- containment vacuum breakers
- containment vacuum breaker isolation valves
- inverters in the uninterruptible ac power supply
- batteries in the dc power system
- IC heat exchangers
- logic units in the DPS
- DPVs
- GDCS injection valves
- IC condensate return valves
- check valves in the GDCS
- software
- DPS processors
- air-operated scram valve AOV-126
- control rods insertion
- DPS load drivers

19.1.5.4.2 Regulatory Criteria

The staff has considered the results and insights from the high-winds risk assessment with respect to the Commission’s objectives for new reactor designs stated in Section 19.1.1 of this document.

19.1.5.4.3 Staff Evaluation

The NRC’s evaluation reestimated tornado strike frequencies based on tornado occurrence rates more specific for the locations in the United States where ESBWR plants are believed likely to be built in the near future (i.e., the Southeast and Midwest), evaluated the effects of tornado missiles on seismic Category II and RTNSS buildings, evaluated the effects from F5 tornado missiles on Category I and II structures, and reviewed the LOPP event tree to determine whether the systems (and associated support systems and structures housing the systems and support systems) are appropriately credited for tornado strike events. The evaluation found that using site specific tornado strike frequencies had little effect on the CDF. The staff finds reasonable the applicant's conclusion that the expected frequency of a tornado strike resulting in core damage is very low. This is because of the low individual frequency and conditional probabilities associated with a tornado strike actually hitting an ESBWR site, combined with the conditional probability that significant damage is done to the plant such that core damage would occur. An important assumption in the risk assessment is that the systems credited for providing core cooling actually will survive a tornado strike. The design of structures important to assuring continued cooling to the core appears to be robust with respect to tornado winds and tornado missiles.

The NRC's evaluation considered the appropriateness of GEH's hurricane strike frequency estimates for coastal sites; evaluated the assumption that seismic Category I, seismic Category II, and RTNSS structures are undamaged by hurricanes; reviewed the LOPP event tree to determine whether the systems (and associated support systems and structures housing the systems and support systems) are appropriately credited for hurricane strike events; and reviewed the assumption that an operating plant would be able to restore ac power within 24 hours for Category 4 and 5 hurricanes.

The evaluation found that the effect on CDF of using generic hurricane occurrence frequencies (i.e., events where a nuclear power plant lost offsite power) was conservative for some sites and potentially optimistic for others. In order for the ESBWR assessment to remain valid when the DC is complete, the systems credited in the PRA and their support systems must be housed in as-built structures that meet the implicit and explicit capacity assumptions made in the PRA regarding the ability of the structures to withstand tornado and hurricane winds and missiles.

19.1.5.4.3.1 Tornado Hazard Frequency

The method of determining tornado frequency used in the risk assessment is conservative for locations west of the Rocky Mountains but potentially nonconservative for locations east of the Rocky Mountains, particularly in the Midwest and the Southeast. The NRC used NOAA data to determine tornado frequency by state, including those areas more likely to experience a tornado (e.g., Oklahoma) or more likely to have an ESBWR built (i.e., announced sites in the Midwest and the Southeast). The NRC finds that the effect on expected CDF of using the higher, more specific occurrence rates should be minimal for most sites (i.e., less than a factor of 4) given the overall low estimated CDF and risk from tornado strikes. Based on this, the staff finds the method used by GEH for determining tornado frequency acceptable.

19.1.5.4.3.2 Evaluation of the Effects of Tornado Strikes

The ESBWR high-winds risk analysis makes the following assumptions regarding tornadoes:

- The event tree initiator used in the risk assessment for a tornado strike on the site is LOPP.
- Offsite power (i.e., preferred power) is lost and is unrecoverable for an F2 to F5 tornado.
- The Fujita Scale and the enhanced Fujita Scale are used interchangeably.

- All site structures can withstand the winds associated with F0 and F1 tornadoes. The only impact on the site is assumed to be LOPP with no additional equipment failures associated with the tornado. These losses of offsite power are assumed to be subsumed in the LOPP frequency evaluated in the internal events risk assessment.

- For windspeeds exceeding approximately 110 miles per hour but less than approximately 195 miles per hour (i.e., F2 and F3 tornadoes as well as Category 3, Category 4, and Category 5 hurricanes), all equipment not located in RTNSS, seismic Category I, or seismic Category II structures is assumed to fail. Equipment in RTNSS, seismic Category I, and seismic Category II structures is credited as available and functional for such high-wind events.

- The risk assessment assumes that within 24 hours of the initiating event, water can be pumped back over the core for gravity injection using pumps (on or off site) powered from onsite or offsite sources. In RAI 19.1-168, the staff requested clarification as to why 24 hours is an appropriate assumption when evaluating a risk assessment for F4 and F5 tornadoes. **RAI 19.1-168 is being tracked as an open item.**

- For windspeeds exceeding approximately 195 miles per hour (i.e., F4 and F5 tornadoes), all equipment in RTNSS buildings is assumed to fail as well as all equipment located in nonseismic, non-RTNSS structures. Only equipment in seismic Category I and II structures is credited throughout all high-winds events. That is, the risk assessment assumes that seismic Category I and II buildings are so robust that they and their contents will remain functional following an F5 tornado strike and equipment inside will not be damaged such that core damage would occur. Equipment inside these structures will fail under these circumstances only because of random failure or loss of support systems that are not entirely contained in seismic Category I or Category II buildings.

The risk assessment estimates shutdown core damage and release frequencies by inserting the shutdown mode tornado strike frequencies into the internal events shutdown LOPP event tree, with the same modeling discussed previously. Tornado and hurricane scenarios during Mode 5, Mode 5 (Open), and Mode 6 (Unflooded) are explicitly quantified. There are four plant operational states during Modes 5 and 6:

(1) Mode 5 is the portion of the TS-defined Mode 5 with the UDW head still in place.

(2) Mode 5 (Open) is the portion of the TS-defined Mode 5 and Mode 6 when the reactor vessel head is still on, and there is no intact containment.

(3) Mode 6 (Unflooded) is the portion of the TS-defined Mode 6 when the reactor head is assumed fully removed, and the reactor well is not flooded. The reactor vessel is open to the reactor building.

(4) Mode 6 (Flooded) is the portion of the TS-defined Mode 6 when reactor well flooding is completed.

Scenarios during Mode 6 (Flooded) are not explicitly quantified as a result of the time assumed available in the risk assessment for recovery because of the extended core cooling capabilities in that mode.

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.1.5.4.3.3 Evaluation of the Effects of Hurricane Strikes

The ESBWR high-winds risk analysis makes the following assumptions regarding hurricanes:

- It is appropriate to model the frequency of significant hurricane strikes by evaluating the history of hurricane strikes at coastal plants that resulted in a loss of offsite power.
- The event tree initiator used in the risk assessment for a hurricane strike on the site is LOPP.
- Offsite power (preferred power) is lost and is unrecoverable for Category 3 to 5 hurricanes.
- All structures can withstand the winds associated with Category 1 and 2 hurricanes. The only impact on the site is assumed to be LOPP with no additional equipment failures associated with the hurricane. The frequency of such losses of power is assumed to be subsumed in the risk assessment's treatment of LOPP for internal events.
- For windspeeds exceeding approximately 110 miles per hour but less than approximately 195 miles per hour (i.e., the windspeeds assumed in the risk assessment for Category 3, 4, and 5 hurricanes), the risk assessment assumes the failure of all equipment not located in RTNSS structures or seismic Category I or seismic Category II structures.
- The risk assessment assumes that within 24 hours of any hurricane strike, water can be provided to cool the reactor from onsite or offsite sources.

The NRC independently evaluated the method used by GEH to estimate hurricane strike frequency at ESBWR coastal sites. The estimate of 1.2×10^{-2}/yr appears to be reasonable for loss of power at most coastal sites, given that (1) not all hurricanes in the vicinity of a nuclear power plant cause loss of offsite power, (2) hurricane windspeeds can be much lower as the distance from the hurricane eye increases, and (3) nuclear power plant sites often have incoming power that is provided by the grid from various directions and regional areas. Loss of offsite power from hurricanes often is the result of salt spray causing arcing of the high-voltage lines (e.g., at the Millstone site). However, at sites such as Turkey Point, when a Category 4 hurricane passed directly over the plant, offsite power was lost for about a week, and it was impossible to access the site for several days because of debris covering the access road(s). The return period of Category 4 hurricanes appears to vary between 16 and 470 years for sites between Florida and Maryland. The return period for Category 5 hurricanes varies from 33 to 500 years for the same sites.

The risk assessment assumes that it is impossible for hurricane winds to significantly damage equipment in seismic Category I, seismic Category II, or RTNSS buildings in a manner that causes core damage. That is, the assessment does not consider that design flaws or construction errors might lead to weaknesses in the as-built design that might make the design vulnerable to such tornado missiles or winds. The NRC is awaiting clarification as to how the risk assessment treated tornado missiles for seismic Category II and RTNSS structures. **RAI 19.1-167 is being tracked as an open item.**

The NRC considers the assumption that it will take the utility less than 24 hours to restore the ability to pump water to the tanks over the core to be optimistic for sites subject to potential Category 4 and 5 hurricanes. Experience at Turkey Point demonstrated the potential for extended loss of offsite power, in combination with the plant site's isolation for many days because of significant debris on roads leading to the site. The NRC is awaiting clarification of the use of the 24-hour assumption. **RAI 19.1-168 is being tracked as an open item.**

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.1.5.4.3.4 Risk Assessment Limitations

The risk assessment has several limitations.

The risk assessment did not evaluate the effect of damage from a hurricane or tornado strike to unprotected equipment in the open (such as fire hydrants). The NRC is awaiting an explanation of the effect on CDF and risk from damage to fire hydrants and other equipment associated with systems credited in the risk assessment. **RAI 19.1-168 is being tracked as an open item.**

The risk assessment assumes that it is impossible for tornado winds or tornado missiles to significantly damage equipment in seismic Category I or Category II buildings in a manner that can cause core damage. The assessment does not consider that design flaws or construction errors might lead to weaknesses in the as-built design that might make the design vulnerable to such tornado missiles or winds. The assessment assumes lack of damage potential for seismic Category II buildings even though they are not designed to withstand tornado missiles (while seismic Category I buildings are specifically designed to be able to withstand tornado missiles). The NRC is also awaiting clarification as to why the risk assessment deemed negligible the possibility of high winds damaging seismic Category I and II structures. **RAI 19.1-169 is being tracked as an open item.**

The risk assessment does not appear to address the effect on RTNSS structures from tornado missiles (including tornado missiles from F2 and F3 tornadoes) even though RTNSS structures are not specifically designed to withstand tornado winds or missiles, but are designed for the maximum windspeed (195 miles per hour) assumed in the risk assessment for Category 5 hurricanes. The NRC is awaiting clarification of how the risk assessment included the effects of these tornado missiles. **RAI 19.1-167 is being tracked as an open item.**

The risk assessment takes credit for systems providing long-term heat removal from the core but does not provide sufficient information on the structures that house these systems and their support systems. The NRC is awaiting information from GEH on which structures house specific equipment important to assuring long-term core cooling. **RAI 19.1-166 is being tracked as an open item.**

Rather than perform an uncertainty analyses for tornado strike events, the risk assessment includes a sensitivity study that increases the expected frequency of tornado occurrences by a factor of 10 with a corresponding increase in expected CDF of approximately a factor of 10. The sensitivity study provides no important additional insights.

The risk assessment does not provide a sensitivity study for hurricane frequency or windspeed. An increase in the expected frequency of hurricanes (e.g., a factor of 10) would make hurricanes the dominant contributor to CDF. The NRC is awaiting clarification as to why the applicant did not perform a sensitivity study for hurricanes. **RAI 19.1-165 is being tracked as an open item.**

In the internal events shutdown PRA, the success of the GDCS was modeled with the assumption that all eight DPVs would be operable and would automatically open. Based on its review of the ESBWR TS, the staff finds that there is no requirement for the DPVs to automatically open, and there is no requirement for the DPVs or the SRVs to be operable in Modes 5 and 6. Instead, there is only a TS surveillance requirement to have a proper vent path for GDCS operability. The TS does not specify the size of this vent path or the number of valves. Therefore, the operator must make the determination of an adequate size vent path. The PRA does not model the failure of the operator to determine adequate vent path size and maintain the proper GDCS path size. Conventional human reliability analysis (HRA) methodologies cannot model this operator error. In response to the staff's RAIs on this issue, GEH recognized the need to maintain the operability of the ADS (both the valves and actuation signals) to support GDCS venting during shutdown. GEH is developing an update to the ESBWR TS to address this issue. Pending staff review of the proposed TS, **RAI 19.1-4, Supplement 1 is being tracked as an open item.** Since the shutdown internal events PRA is used for the shutdown high-winds risk assessment, this issue affects the results.

19.1.5.4.4 Conclusions

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.1.6 Safety Insights from the Probabilistic Risk Assessment for Other Modes of Operation

19.1.6.1 <u>Results and Insights from Internal Events Low-Power and Shutdown Operations Probabilistic Risk Assessment</u>

19.1.6.1.1 Summary of Technical Information

The staff's review of the ESBWR internal events shutdown PRA is based on the results reported in Chapter 16, "Shutdown Risk." The evaluation encompasses plant operation in cold shutdown and refueling modes, as discussed in TS Modes 5 and 6. Mode 5 begins when the reactor coolant temperature in the RCS drops to or below 93.3 °C (200 °F) while the plant is cooling/shutting down. Mode 6 begins when one or more of the reactor vessel head closure bolts is less than fully tensioned. This risk analysis addresses conditions for which there is fuel in the RPV.

GEH did not quantitatively evaluate operation in Mode 4 (i.e., stable shutdown in which the RCS temperature is less than XXX°C (420 °F) and greater than 93.3 °C (200 °F)). In this mode, the reactor mode switch is in the shutdown position and control rod insertion is completed. The initial RPV conditions (pressure and temperature) for Mode 4 are the same as power-operating values.

The staff reviewed the TSs and concluded that all credited systems in the PRA have the same TS for Modes 1 through 4, except for containment. Mode 4 requires containment integrity, but the containment is deinerted, limiting the plant's ability to control hydrogen generation following a severe accident. The duration of this mode is assumed to be 8 hours. The applicant stated that the CDF contribution of Mode 4 is bounded by the full-power PRA; Chapter 8.1.4 of the ESBWR full-power PRA assesses the LRF contribution from this mode. The staff finds this PRA modeling to be acceptable.

The scope of the shutdown PRA is that of a Level 1 PRA. The different accident sequences are characterized according to whether the core is damaged or not. The PRA did not define core damage. In RAI 19.1-196, the staff requested additional information documenting the success criteria used in the shutdown PRA. **RAI 19.1-96 is being tracked as an open item.**

In RAI 5.4-59, the staff raised questions regarding the capability of the RWCU/SDC system to operate during Modes 5 and 6. The staff requested that the DCD discuss the normal vessel levels for RWCU/SDC operation in all modes, including Modes 5 and 6. The staff also requested calculations demonstrating under what temperatures and levels the RWCU/SDC systems can adequately remove decay heat in Modes 4, 5, and 6 (with the RPV head installed), including any minimum and maximum levels. In addition, the staff asked the applicant to explain the RWCU/SDC flow and mixing within the vessel and within the shroud. **RAI 5.4-59 is being tracked as an open item.**

The critical safety functions essential to the shutdown model are decay heat removal and inventory control. Containment is assumed to be open. The TS in Modes 5 and 6 does not require containment integrity for severe accident mitigation.

GEH did not quantitatively assess the safety functions of spent fuel cooling and reactivity control. The applicant stated that the spent fuel cooling function will be maintained during shutdown modes just as it will be during full-power modes. The applicant assumed this function to have no significant impact on the shutdown model. Regarding reactivity control, all control rods are fully inserted for the duration of the modeled modes, and therefore, ATWS is not an issue. The DCD addresses reactivity control during shutdown deterministically.

The applicant assessed the following four plant operational states during Modes 5 and 6: Mode 5 Mode 5 (Open), Mode 6 (Unflooded), and Mode 6 (Flooded), as previously defined.

19.1.6.1.2 Significant Accident Sequences and Leading Contributors

The applicant estimated the mean ESBWR shutdown CDF from internal events to be 9.4×10^{-9}/yr. This estimate is not considered to be the true expected CDF. The very low CDF represents the applicant's effort to reduce or eliminate the contributors to core damage found in previous PRAs through improvements in plant design. Areas of shutdown risk where modeling is least complete or nonexistent (such as operator errors of commission and rare/new initiating events) could become important contributors to risk.

The ESBWR shutdown CDF accounts for the duration (in hours) of operation in Modes 5 and 6 per outage and the anticipated calendar outage frequency of one refueling outage every 2 years. Therefore, the staff believes the shutdown CDF can be added to the full-power internal events CDF.

All evaluated shutdown core damage events are assumed to result in a large release because of the potential for the containment to be open during the outage. CCFP is not affected because the containment is not being used as a mitigating system during shutdown. Thus, the applicant reported the shutdown LRF from internal events to be 9.4^{-9}/yr.

RWCU/SDC drainline breaks below TAF and instrument line breaks below TAF that may occur during each of the four plant operational states (Mode 5, Mode 5 (open), Mode 6 (unflooded), and Mode 6 (flooded)) comprise over 90 percent of the ESBWR internal events shutdown CDF. It is necessary to flood the drywell and the vessel up to a level above TAF to reach a safe core-cooling condition for those breaks below TAF.

The total contribution to the ESBWR internal events shutdown CDF from other shutdown initiating events, such as an LOPP, loss of PSW, and loss of RWCU/SDC, is less than 10 percent.

19.1.6.1.3 Regulatory Criteria

The staff used the risk insights gained from SECY-97-168, “Issuance for Public Comment of Proposed Rulemaking Package for Shutdown and Fuel Storage Pool Operation,” issued July 1997, and guidance provided in the associated SRM. The staff considered the results and insights for shutdown risk assessment with respect to the Commission’s objectives for new reactor designs stated in Section 19.1.1 of this report.

19.1.6.1.4 Staff Evaluation

Based on key risk insights from SECY-97-168, NUMARC 91-06, “Guidelines for Industry Actions to Assess Shutdown Management,” issued December 1991, and previous shutdown PRAs, the staff’s review of the ESBWR shutdown PRA considered shutdown TSs, critical operator actions, and proposed regulatory oversight for non-safety systems identified by the RTNSS process. In SECY-97-168, the staff concluded that the current level of shutdown safety was achieved through the use of voluntary measures (including those identified in NUMARC 91-06). However, GEH did not identify outage planning and control consistent with NUMARC 91-06 as a key risk insight in Table 19.2-3, “Risk Insights and Assumptions,” of DCD Tier 2, Revision 4. In RAI 19.1-149, Supplement 1, the staff requested GEH to address this issue. **RAI 19.1-149, Supplement 1 is being tracked as an open item.**

As discussed previously, RWCU/SDC drainline breaks below TAF and instrument line breaks below TAF that may occur during each of the four plant operational states (Mode 5, Mode 5 (Open), Mode 6 (Unflooded), and Mode 6 (Flooded)) comprise over 90 percent of the ESBWR internals event shutdown CDF. The LDW is equipped with a personnel hatch and an equipment hatch to allow access to the containment. These hatches are closed during normal operation but may be open during refueling. Closure of these two hatches is required for successful drywell flooding and to prevent core damage following a break below TAF. Manual closure of the LDW hatches is a risk-significant operator recovery action.

The RTNSS program includes closure of the LDW hatches, which is described in the Availabilities Control Manual (ACM). The ability to close the hatch is covered during Modes 5 and 6. Immediate action is required if hatch closure is unavailable for any reason. Availability control Limiting Condition for Operation (ACLCO) 3.6.2 in the ACM states, “The lower drywell personnel air lock and lower drywell equipment hatch shall be AVAILABLE for closure.” It is stated in ACLCO 3.6.2 that the ACLCO is applicable in Modes 5 and 6. However, the PRA states that the ACLCO is required in Modes 5 and 6 only during operations with the potential to drain the vessel. This statement is inconsistent with ACLCO 3.6.2. In RAI 19.1-123, Supplement 2, the staff requested the applicant to address this inconsistency. **RAI 19.1-123, Supplement 2 is being tracked as an open item.**

The staff noted that Availability Control Surveillance Requirement (ACSR) 3.6.2.2 and 3.6.2.3, whose purpose is to verify—with a frequency of 30 days—that, during an outage, the LDW equipment hatch and personnel airlock can be secured, is inconsistent with NUMARC 91-06 guidance and operating experience. GEH responded that the intent of Availability Control (AC) 3.6.2 is to allow the licensee to mitigate the effects of a pipe break in a line from the vessel below TAF. AC 3.6.2 provides administrative controls that allow the licensee to establish a boundary to flood the LDW to above the level of the break, thus ensuring that the fuel in the core is covered with water.

GEH stated that this guidance is not intended to satisfy NUMARC 91-06 recommendations for preventing fission product release from containment. The staff believes that this guidance should satisfy NUMARC 91-06 recommendations for the following reason. For this new design, to mitigate pipe breaks below TAF, which comprise over 90 percent of the shutdown CDF, containment closure is necessary to prevent core damage that would result in fission product release from containment if not performed. The ACSR frequency of 30 days is most likely longer than the outage itself. The NUMARC 91-06 guidelines state, “a procedure should be established to assure that closure can be accomplished in a time commensurate with plant conditions,” recognizing that conditions change during the outage. In RAI 19.1-123, Supplement 2, the staff requested GEH to address this issue. **RAI 19.1-123, Supplement 2 is being tracked as an open item.**

Once a postulated LOCA has been detected, the plant operator must correctly diagnose the situation, make the decision to close the hatches, gain access to the -6400 millimeter (-21 foot) in the reactor building, and manually close the equipment hatch and the personnel air lock. Two key assumptions substantiate the human reliability estimates (1) outage personnel will be continuously located in the area of the doors, and (2) closure of both the equipment hatch and personnel hatch can be done from outside the LDW/containment. GEH did not recognize item 1 as a key risk insight in Table 19.2-3 of DCD Tier 2, Revision 4. The ability to close the equipment and personnel hatch from the outside is a key design feature necessary to support hatch closure reliability estimates. GEH did not document this design insight as a key risk insight in Table 19.2-3 of DCD Tier 2, Revision 4. In RAI 19.1-4, Supplement 1, the staff requested GEH to address this issue. **RAI 19.1-4, Supplement 1 is being tracked as an open item.**

To mitigate LOCAs in Mode 5 and Mode 5 (Open), with the exception of feedwater line breaks, four DPVs must open for the GDCS to function. The shutdown PRA modeled the success of the GDCS assuming that all eight DPVs would be operable and would automatically open. Based on its review of the ESBWR TS, the staff found that there is no requirement for the DPVs to automatically open, and there is no requirement for the DPVs or the SRVs to be operable in Modes 5 and 6. Instead, there is only a TS surveillance requirement to have a proper vent path for GDCS operability. The TS does not specify the size of this vent path or the number of valves. Therefore, the operator must determine an adequate size vent path.

The PRA does not model the failure of the operator to determine adequate vent path size and to maintain the proper GDCS path size. This operator error cannot be modeled with conventional HRA methodologies. In response to the staff’s RAIs on this issue (i.e., RAIs 19.1-93, 19.1-94, 19.1-95, and 19.1-143), GEH recognizes the need to maintain ADS (both the valves and actuation signals) operability to support GDCS venting during shutdown. GEH is updating the

ESBWR TSs to address this issue. **RAIs 19.1-93, 19.1-94, 19.1-95, and 19.1-143 are being tracked as open items.** Since the shutdown internal events PRA is also used for the high-winds, fire, and flood risk assessments, this issue affects the results of those assessments.

As shown in Table 16.6.3 of the ESBWR shutdown PRA, the top eight dominant cutsets contribute over 92 percent of the risk. These eight cutsets initiate by an RWCU/SDC drainline LOCA below TAF or an instrument line LOCA below TAF in each of the four plant operational states—Mode 5, Mode 5 (Open), Mode 6 (Unflooded), and Mode 6 (Flooded). Subsequent failure of the operator to close the drywell hatch leads to failure of drywell flooding and core damage.

In Section 19.2.4.2 of the DCD, GEH stated that it judged the offsite consequences from shutdown risk to be negligible since significant shutdown events occur during Mode 6, which does not begin until approximately 96 hours after shutdown. In RAI 19.1-159, the staff requested that GEH revise this statement based on two assumptions. In Chapter 16 of the PRA, over 40 percent of the internal shutdown CDF occurs in Mode 5. Furthermore, NUREG/CR 6595, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," issued January 1999, states on page 4-3, "The results indicate that source terms which involve a release of about 10% or less of the core iodine inventory (10% iodine releases are associated with early fatalities in accidents that occur at full-power), offsite doses generally fall below the early fatality threshold approximately 8 days or less after shutdown." Based on these assumptions, the consequences of a shutdown severe accident occurring during Modes 5 and 6 approximately 8 days or less after shutdown is not negligible. **RAI 19.1-159 is being tracked as an open item.**

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.1.6.1.5 Risk-Significant Functions/Features, Phenomena/Challenges, and Human Actions

Listed below are key ESBWR design features that significantly reduce the shutdown CDF as compared to the CDF for operating BWR designs. These design features are described below by initiating event category.

19.1.6.1.5.1 Operator-Induced Draindowns/Loss-of-Coolant Accidents

The ESBWR design has reduced the number of potential RPV drain pathways caused by postulated system misalignment during shutdown conditions. As compared to residual heat removal systems in current BWRs, the RWCU/SDC system in the ESBWR does not have the potential to divert RPV inventory to the suppression pool through the suppression pool suction, return, or spray lines. The RWCU/SDC system does not provide any drywell spray function, so the potential for draining the RPV through drywell spray does not exist. In addition, the applicant eliminated recirculation lines in the ESBWR design, further reducing potential RPV drain paths.

The RWCU/SDC system is the only operating system that has the potential to drain the RPV during Modes 5 and 6. This system is connected to the RPV during shutdown, and it is used to discharge excess reactor coolant to the main condenser or to the radwaste system during startup, shutdown, and hot standby conditions.

The RWCU/SDC system containment penetrations have redundant and automatic power-operated containment isolation valves that close upon signals from the leakage detection and isolation system in Modes 5 and 6. In Modes 5 and 6, TS 3.3.6.3 and 3.3.6.4 require the RWCU/SDC system and the FAPCS containment isolation valves to close on low reactor vessel water level (level 2). This risk-significant TS provides protection against postulated breaks in the RWCU/SDC system outside containment.

To eliminate the use of freeze seals for isolating lines attached to the vessel, all power-operated equipment and valves that require maintenance have maintenance valves installed to preclude the need for freeze seals. Lines to the low-conductivity waste system have been relocated to a point downstream of the isolation valves. The leak detection and isolation system monitors the lines to the primary sampling system, which have two redundant isolation valves. The only piping penetrations that are not downstream of the isolation valves are 20-millimeter drain and vent lines near the isolation valves. These small lines have two normally closed manual valves and a threaded cap. Based on these design insights, GEH did not quantitatively assess operator-induced losses of reactor vessel inventory. The staff accepts that GEH did not quantitatively evaluate operator-induced loss of reactor vessel inventory. However, GEH did not recognize these design insights as key risk insights and did not document them in Table 19.2-3 of DCD Tier 2, Revision 4. In RAI 19.1-4, Supplement 1, the staff requested GEH to address this issue. **RAI 19.1-4, Supplement 1 is being tracked as an open item.**

To reduce the likelihood of the reactor vessel inventory being drained into the feedwater lines, the RWCU/SDC lines returning to the feedwater lines are each provided with redundant check valves in series located in the main steam tunnel. A single, power-operated isolation valve in each line is located upstream of the check valves and inside the reactor building. The FAPCS and CRDS connections are downstream of the two check valves. A postulated break in the RWCU/SDC piping system inside the reactor building, which would otherwise allow reactor coolant to flow backwards through the main feedwater lines and spill into the reactor building, will be isolated by either redundant RWCU/SDC check valves or feedwater check valves, even if a single failure of one check valve were to be assumed.

GEH evaluated the draining of the RPV during FMCRD maintenance but did not consider it to be a shutdown PRA initiating event. If the operator were to inadvertently remove the control rod after the FMCRD is out without first installing the temporary blind flange, or conversely, if the operator were to inadvertently remove the FMCRD after first removing the control rod, an unisolable opening in the bottom of the reactor would be created, resulting in drainage of reactor water. The possibility of inadvertent reactor draindown by this means is considered remote for the following reasons:

- Procedural controls similar to those of current BWRs provide the primary means for prevention. Current BWR operating experience demonstrates the acceptability of this approach. There has been no instance of an inadvertent draindown of reactor water caused by simultaneous CRD and control rod removal.

- During drive removal operations, personnel are required to monitor under the RPV for water leakage out of the CRD housing. Abnormal or excessive leakage occurring after a partial lowering of the FMCRD within its housing indicates the absence of the full metal-to-metal seal between the control rod and control rod guide tube required for full drive removal. In this event, the FMCRD can then be raised back into its installed position to stop the leakage and allow corrective action.

In the PRA, GEH stated that the COL applicant will develop maintenance procedures with provisions to prohibit coincident removal of the control rod and CRD of the same assembly. In addition, the COL applicant will develop contingency procedures to provide core and spent fuel cooling capability and mitigative actions during CRD replacement with fuel in the vessel. However, the staff noted that GEH did not capture these insights in Table 19.2-3 of DCD Tier 2, Revision 4. In RAI 19.1-4, Supplement 1, the staff requested GEH to address this issue. **RAI 19.1-4, Supplement 1 is being tracked as an open item.**

Should a LOCA or an operator-induced loss of inventory occur and all active, non-safety-related systems be unavailable, or, if the operator fails to initiate injection after successful manual RPV depressurization, the passive GDCS will automatically inject water into the RPV.

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.1.6.1.5.2 Loss of Both Operating Reactor Water Cleanup/Shutdown Cooling Trains

At the beginning of every shutdown outage, both RWCU/SDC trains are assumed to be running, with the pumps varying their speed to meet the cooldown rate objectives. The shutdown PRA also assumes that both trains are running during Modes 5 and 6; however, only one train is required to prevent reactor coolant boiling. More importantly, the focused PRA results, which were used to identify non-safety-related systems for RTNSS, were based on the assumption that both trains of the RWCU/SDC system are running until the reactor cavity is flooded. To ensure that the focused PRA results remain valid, operation of both trains of the RWCU/SDC system during Modes 5 and 6 is an important risk insight which GEH did not capture in Table 19.2-3 of DCD Tier 2, Revision 4. In RAI 19.1-4, Supplement 1, the staff requested GEH to address this issue. **RAI 19.1-4, Supplement 1 is being tracked as an open item.**

The RWCU/SDC function may fail for any of the following reasons:

- failure of both RWCU/SDC trains
- isolation of the RWCU/SDC system caused by RPV low-level or leakage detection and isolation system signals
- LOPP
- loss of RCCWS or PSWS

Should any of these scenarios occur, the ESBWR can be cooled via the ICs which offer an alternative, automated, passive, core-cooling path not available in current operating BWRs.

TSs 3.3.5.3 and 3.3.5.4 require the ICS to be operable in Mode 5. The ICS automatically initiates upon high reactor vessel steam dome pressure, low reactor vessel water-level 2, and low-low reactor vessel water-level 1. In RAI 19.1-144, Supplement 1, the staff raised a question regarding the effect of noncondensable gases on ICS performance during Mode 5. **RAI 19.1-144, Supplement 1 is being tracked as an open item.**

Should the ICS fail, three FAPCS functions (coolant injection, suppression pool cooling, and backup SDC) are included within the scope of RTNSS at shutdown. In the unlikely event that these functions fail, the ESBWR design has a second, automated, passive core cooling path via the GDCS. The GDCS is required to be operable and automatically initiates upon reactor vessel water level- (level 1), during Modes 5 and 6, except when the new fuel pool gate is open and the water level exceeds 7.01 meters (23.0 feet) over the top of the RPV flange. Adequate venting for the GDCS during shutdown is discussed in Section 19.1.6.1.4 of this report.

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.1.6.1.6 Insights from Uncertainty, Importance, and Sensitivity Analyses

The staff used the results of the applicant's importance analyses to identify (1) SSCs and/or human actions whose reported reliability contribute most to achieving the low reported shutdown CDF (RAW), and (2) SSCs and/or human actions whose reported reliability contribute most to a reduction in shutdown CDF if the reliabilities were improved (risk reduction worth).

Since the reported ESBWR shutdown CDF is very low and clearly meets the Commission's safety goals and the EPRI ALWR CDF requirements, the staff focused on the results of the GEH risk achievement analyses. The staff used these results to identify (1) the SSCs for which it is particularly important to maintain the reliability/availability levels assumed in the PRA (e.g., by testing and maintenance) to avoid significant increases in CDF, and (2) the human actions that, if they were to fail, would have the largest impact on the shutdown PRA.

GEH performed risk importance analyses at the component/human action/initiating event level. It is important to note that the PRA did not assess BOCs; therefore, they are not included in the importance analyses. BOCs can originate only in the ICS, RWCU/SDC system, FAPCS piping, or instrument lines, which are the only systems that remove reactor coolant from the containment during shutdown. The rest of the RPV vessel piping is isolated. The RWCU/SDC system, FAPCS, and ICS containment penetrations have redundant and automatic power-operated, safety-related containment isolation valves that close upon signals from the leakage detection and isolation system in Modes 5 and 6.

The high reliability of the leakage detection and isolation system provides the basis for the screening of (1) shutdown LOCAs outside of containment and (2) operator-induced losses of reactor vessel inventory during shutdown. Therefore, the high reliability of the leakage detection and isolation systems is a key risk assumption, but GEH did not document it as a key risk insight in Table 19.2-3 of DCD Tier 2, Revision 4. In RAI 19.1-4, Supplement 1, the staff requested GEH to address this issue. **RAI 19.1-4, Supplement 1 is being tracked as an open item.**

The other major risk insight from the GEH risk achievement analyses involves the occurrence of an RWCU/SDC LOCA or an instrument line LOCA. These LOCAs below TAF have RAW values exceeding 1×10^{-6} and comprise over 90 percent of the shutdown CDF/LRF. To prevent core damage, the operator must close the drywell hatch.

The occurrence of a loss of PSW or RWCU/SDC during Mode 6 when the cavity is unflooded is also risk significant and has RAW values of 3.7×10^{3} and 479, respectively. During Mode 6 with the vessel head removed, the ICS is not available for DHR.

GEH also performed a number of sensitivity studies to gain insights about the impact of uncertainties on the reported shutdown CDF. Specifically, these studies show how sensitive the shutdown CDF is to potential biases in numerical estimates assigned to initiating event frequencies, equipment unavailabilities, and human error probabilities.

Similar to the full-power analysis, GEH performed two separate analyses to investigate the impact of shutdown operation without credit for non-safety-related defense-in-depth systems. The focused PRA sensitivity study evaluates whether passive systems alone are adequate to meet the Commission's safety goals of less than 1×10^{-4}/yr for CDF and less than 1×10^{-6}/yr for LRF. The focused PRA retains the same initiating event frequencies as the baseline PRA and sets the status of non-safety-related systems to failed, while safety-related systems remain unchanged in the model.

19.1.6.1.6.1 Risk Impact of Non-safety-Related Systems (Focused PRA)

The intent of the focused PRA is to determine the impact to CDF and LRF caused by removing credit for non-safety systems. The results are then compared to the following NRC criteria to determine whether systems should be considered for RTNSS:

- CDF less than 1×10^{-4}/yr
- LRF less than 1×10^{-6}/yr

GEH performed focused PRA analyses for the following shutdown PRA models:

- internal
- fire
- flood
- high winds

The shutdown analyses do not require evaluation of LRF because the containment is assumed to be open.

The following systems are assumed to be unavailable for the focused analyses: emergency diesel generators, condenser, condensate and feedwater, CRD injection and FMCRD, FAPCS, RWCU/SDC, FPS injection, DPS, MSIV, reactor component cooling water (RCCW), TCCW, plant air, nitrogen, PSW, FMCRD groups, and PIP buses A3 and B3.

ESBWR PRA Table 20.2-1, "PRA Focus Results," shows the results of the focused PRA analyses with and without RTNSS. The results of the focus PRA for shutdown internal events and the focused PRA for shutdown fire events do not meet the NRC criteria (CDF/LRF) without RTNSS credited. However, the CDF and LRF criteria are met if the RTNSS is credited as shown. These RTNSS systems/operator actions for Modes 5 and 6 include, for example, closure of the LDW hatches, availability of two FAPCS trains, and availability of two standby diesel generators. The ACM lists the availability controls for these systems.

19.1.6.1.6.2 Loss-of-Coolant Accident Frequency

Because of the lower temperatures and pressures in the RPV during shutdown, GEH applied a reduction factor to the LOCA frequencies for the shutdown PRA. Section 16.3.1.2.1 of the ESBWR shutdown PRA documents the GEH basis for the reduction. This sensitivity case shows the following results with no reduction factor applied.

- baseline results = 9.37×10^{-9}/yr

- sensitivity results = 8.69×10^{-8}/yr

This case results in an order of magnitude increase in CDF. The results are to be expected since the CDF contribution from LOCA sequences comprise over 90 percent of the baseline result. Therefore, the baseline ESBWR shutdown CDF/LRF results depend directly on the assumed LOCA frequencies. Without the reduction factor, the ESBWR shutdown CDF results are still below the NRC safety goals.

19.1.6.1.6.3 Lower Drywell Hatch Sensitivity

RWCU/SDC drainline breaks below TAF and instrument line breaks below TAF that may occur during all four plant operational states comprise over 90 percent of the ESBWR internals event shutdown CDF. For the breaks below TAF, it is necessary to flood the drywell and the vessel up to a level above the TAF to reach a safe core-cooling condition. Failure to close the LDW equipment hatch and the personnel air lock following a postulated LDW LOCA is assumed to lead to core damage.

The PRA evaluates two hatch closure events. For instrument line LOCAs, GEH estimated that 360 minutes would be available to close the hatch. For RWCU drainline breaks, GEH estimated that 90 minutes would be available. Both times are based on the worst-case pipe break scenario.

The baseline case used screening values for the operator action to close the hatch. A failure probability of 0.01 was applied to the case in which 360 minutes would be available for the action. A failure probability of 0.1 was applied to the case in which 90 minutes would be available.

GEH also ran a sensitivity case applying a 50-percent failure rate for both hatch closure events. The resulting CDF is 3.41×10^{-7}/yr. The resulting ESBWR shutdown CDF and LRF are more than a magnitude higher than the baseline CDF.

GEH ran a sensitivity case assuming that no LDW entry is allowed until Mode 6. This eliminates the Mode 5 and Mode 5 (Open) sequences that include drywell hatch closure. The ESBWR shutdown CDF and LRF that result are approximately half the baseline value of 5.53×10^{-9} /yr.

As with the LOCA frequencies, any change to the hatch closure failure rates will result in a proportional change to the ESBWR shutdown CDF. The two hatch closure events are in the top eight cutsets and account for over 90 percent of the total shutdown CDF.

19.1.6.1.6.4 Operator Action Sensitivity

During shutdown, the plant relies on operator actions for accident mitigation more than it does during power operation. Several systems have no automatic actuation and rely on operators to initiate (i.e., FPS, FAPCS, CRD). Human actions are the only barrier between the initiating events and core damage for LOCA events below TAF. The operator must close the equipment and personnel hatches to allow the drywell to flood, which will, prevent core damage.

GEH evaluated the following two operator action sensitivity cases.

Case 1 sets all recovery actions to TRUE (failed). This eliminates several systems from possible accident mitigation because CRD (during shutdown), FAPCS, FPS, and manual depressurization depend completely on human action for initiation. The RWCU/SDC system also requires operator action following LOPP. Most importantly, the operator's ability to close the equipment and personnel hatch following a LOCA was also set to TRUE (failed).

The resulting CDF for Case 1 is 2.29×10^{-6}/yr. Case 1 results show an increase of more than 2 orders of magnitude in CDF over the baseline case. For the LOCA below TAF, with the operator failing to close the equipment and personnel hatch, these sequences go directly to core damage. Therefore, for these initiating events, the CDF value is equal to the initiating event value.

Case 2 assigns all recovery actions a low human error probability of 1×10^{-3}. This human error probability estimate is about 1 order of magnitude lower than most modeled human actions. It shows how CDF could be affected if credit is taken for very effective operator response to transients.

The resulting CDF for Case 2 is 7.78×10^{-10}/yr. Case 2 results in a decrease in CDF of approximately 1 order of magnitude when compared to the base shutdown case. Human errors still dominate the top cutsets in this case. Even with the reduced failure rates, human errors are still generally higher than the common-cause equipment failures that appear in the top cutsets. Based on these sensitivity studies, the staff concludes that the ESBWR shutdown risk is sensitive to human error.

The staff finds the applicants sensitivity studies described above to be acceptable.

19.1.6.1.6.5 Risk Impact of Minimal Technical Specification Compliance

The staff requested that GEH provide a sensitivity study that shows the difference in CDF between the baseline PRA result and a case in which only the systems required to be operable according to the TS credited. This includes the ICS, GDCS, and ADS. GEH estimated the CDF to be less than 5×10^{-7}/yr for the case in which only systems required by the TS are credited as compared to the baseline shutdown CDF which GEH estimated to be less than 1×10^{-8}/yr. The increase is somewhat significant, but still relatively low. The top cutsets in the baseline case, which involve LOCAs below TAF, are unaffected by the change assumed in the study. These sequences require closing of the drywell hatches to prevent core damage. This sensitivity case result is also relatively low since all the systems removed from the model depend on human action. Adequate venting for the GDCS at shutdown impacts the results of this sensitivity case and is discussed in Section 19.1.6.1.4.

19.1.6.1.7 Conclusions

Based on the discussions above, the staff concludes that the methodology and approach of the internal events low-power and shutdown risk analysis are acceptable. Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability of identification of design insights.

19.1.6.2 <u>Results and Insights from External Events Low-Power and Shutdown Operations Probabilistic Risk Assessment</u>

19.1.6.2.1 Significant Accident Sequences and Leading Contributors

Based on the level 1 internal events shutdown PRA, GEH performed a quantitative fire, flood, and high-winds risk analysis. Using the MIN-MAX method, GEH also conducted an SMA. This section briefly summarizes the methodology used to complete each assessment and discusses the significant severe accident sequences and leading contributors.

19.1.6.2.1.1 Regulatory Criteria

The staff considered the results and insights for shutdown risk assessment with respect to the Commission's objectives for new reactor designs stated in Section 19.1.1 of this report.

19.1.6.2.1.2 Shutdown Fire Significant Accident Sequences and Leading Contributors

The ESBWR shutdown fire assessment was performed according to the guidance in NUREG/CR-6850. The guidance does not explicitly cover shutdown conditions, but the following tasks were completed, as applicable:

- Task 1—Plant Boundary and Partitioning
- Task 2—FPRA Component Selection
- Task 3—FPRA Cable Selection
- Task 4—Qualitative Screening
- Task 5—Fire-Induced Risk Model
- Task 6—Fire Ignition Frequencies
- Task 7—Quantitative Screening

As in the full-power fire assessment, GEH conservatively assumed that fires would propagate unmitigated in each fire area and damage all functions in the fire area with a few exceptions. Fire suppression is not credited. During shutdown conditions, a fire barrier may not be intact because of maintenance activities. The shutdown fire analysis assumes that all barriers are intact, or an added fire watch would increase the probability of fire detection and suppression and would also help to restore the fire barrier in time to prevent fire propagation.

The analysis estimated the probability of a fire barrier failure to be 7.4×10^{-3}. To determine whether a failed fire barrier is significant, In RAI 19.1-126, Supplement 1, the staff requested that GEH submit information describing which fire barriers are particularly risk significant and how the COL holder will choose between roving and continuous fire watches for barriers of increased risk significance. **RAI 19.1-126, Supplement 1 is being tracked as an open item.**

The screening criterion for the shutdown fire model is that the postulated fire has to result in one of the initiating events defined in the shutdown internal events PRA model. The critical safety functions essential to the shutdown model are decay heat removal and inventory control. Similar to the baseline shutdown PRA results, all evaluated shutdown fire core damage events are assumed to result in a large release because of the potential for the containment to be open during the outage.

Reactivity control and spent fuel pool cooling are assumed to have no significant impact on the shutdown model. Power availability is modeled for its impact on DHR. Loss of power is evaluated as an initiating event, and the model includes power dependencies for systems. Fire-induced IORV is also not a shutdown fire-initiating event. Line breaks, or a stuck-open relief valve, that occur above the reactor vessel water level (level 3) mark are not initiating events because RWCU/SDC system operation is not expected to be impacted. As discussed in Section 12 of the ESBWR PRA report, GEH developed the shutdown fire ignition frequencies using operating experience data published by the NRC.

The total CDF for all shutdown fire scenarios is 2.7×10^{-8}/yr. In Section 12.1.8.5, of the ESBWR PRA report, GEH stated, "Note for all shutdown fire scenarios, all the operator actions are assumed to be failed for conservatism." However, GEH discusses the impact of operator errors in the results, and the cutsets contain operator errors. To understand MCR fire risk, the staff, in RAI 19.1-129, Supplement 1, requested a sensitivity study that credits only automated equipment or information in the PRA regarding the operator's ability to monitor the RWCU/SDC system status, reactor vessel water level, and RCS pressure from the back panel rooms. The staff also requested an availability control to prevent both remote shutdown panels from being out of service at the same time or administrative controls that would prevent both shutdown panels from being out of service at the same time. **RAI 19.1-129, Supplement 1 is being tracked as an open item.**

In summary, postulated turbine building fires and service water building fires during Mode 6 (Unflooded) operation contribute 93 percent of the shutdown fire CDF since the ICs are unavailable for DHR. The following sections describe the top two sequences contributing approximately 98 percent of the risk, as reported by GEH.

Shutdown Fire Scenario 1, contributing approximately 70 percent, is initiated by a postulated fire in the turbine building general area (F4100 fire area) during Mode 5, Mode 5 (Open), and Mode 6 (Unflooded) operation. This fire is assumed to result in a complete failure of the service air system because of cable failures, which lead to the closure of all RWCU containment isolation valves outside the containment. If the cabling for the instrument air system is designed with separation criteria, GEH indicated that this fire scenario should be screened out. Other systems failed by a postulated fire in area F4100 include the condensate and feedwater system, turbine building secondary closed cooling water system (TCCWS), service air system, and UPS buses in the turbine building.

The cabling for the RCCW system and PSW system is assumed not to be failed by a fire in F4100 since these two systems have been identified as part of the RTNSS program. The design requirements for RTNSS ensure that a postulated fire would not damage both trains.

Shutdown Fire Scenario 2, contributing approximately 28 percent of the CDF, is initiated by a postulated fire in the service water building (fire area F7300). This fire scenario is assumed to result in a complete failure of the PSW system, which results in the loss of DHR. Failure of the PSW system results in the failures of the RCCW, TCCW, and RWCU/SDC systems. GEH indicated that this fire scenario is conservative since the components in this fire area should be well separated into four subareas.

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.1.6.2.1.3 Shutdown Floods Significant Accident Sequences and Leading Contributors

As in the full-power assessment, the shutdown internal flooding analysis is performed using equipment locations based on existing plant layout drawings. Also similar to the full-power assessment, buildings were divided into flood zones based on separation for flooding. Flood zones that do not contain flood sources or PRA equipment were screened from consideration.

Depending on the building and the origin of the flood, GEH considered the following aspects for flood propagation: automatic flood detection systems, automatic systems to terminate flooding, watertight doors to prevent the progression of flooding, sump pumps, and other design or construction characteristics that contribute to minimize the consequences of flooding.

GEH used the internal events shutdown PRA to construct the shutdown flooding PRA. The shutdown PRA uses the same system success criteria, and the containment hatches are assumed to be open. As in the full-power assessment, the initiating event frequency for each flood zone was estimated by summing the frequencies for flood components and piping for the system under consideration. GEH referenced NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," issued February 2007 for the rupture features, and NSAC-60, "Oconee PRA: A Probabilistic Risk Assess of Oconee, Unit 3," issued July 1984 for the expansion joint failure data.

GEH estimated the shutdown CDF for each flood damage state by quantifying the loss of RWCU/SDC for the three plant operating states: Mode 5, Mode 5 (Open), and Mode 6 (Unflooded). The applicant did not consider Mode 6 (Flooded) since the water above the core will be adequate to provide core cooling for 24 hours.

The estimated mean shutdown flooding CDF is 5.2×10^{-9}/yr. The estimated LRF is also 5.2×10^{-9}/yr since the containment is assumed to be open. The estimated CDF accounts for the number of hours in each operating mode and the frequency of an outage (once every 2 years). The following paragraphs describe the top four flooding sequences which contribute approximately 67 percent of the shutdown flooding CDF.

Flooding Sequence 1, contributing about 28 percent, is initiated by floods in containment (Flood Zone E50A_L and Flood Zone E 50D_L), which impact the GDCS.

Flooding Sequence 2, contributing about 21 percent, is initiated by floods in the service water building (Flood Zone SF-P41A_S_SD and Flood Zone SF-P41B_S_SD), which impact the PSW system.

Flood Sequence 3, contributing about 12 percent, is initiated by a flood in the CRD pump room (Flood Zone RBCRD-6400-C12_S_SD).

Flood Sequence 4, contributing about 6 percent, is initiated by floods in the reactor building hydraulic control unit (Flood Zone RBH-11500-C12_S_SD and Flood Zone RBH-11599-G31A_S_SD), which impact the RWCU/SDC system.

GEH assumed that floods in containment during shutdown will not affect ICS components or the DPVs because these components are relatively high in the containment. As a result, the top 200 shutdown flooding cutsets occur during Mode 6 (Unflooded) when the ICS is not available for DHR.

19.1.6.2.1.4 Shutdown High-Winds Significant Accident Sequences and Leading Contributors

As in the full-power assessment, GEH performed the following major steps to complete the high-wind risk analysis:

- tornado hazard frequency
- tornado-induced plant impacts

- calculation of tornado-induced CDFs and release frequencies
- hurricane hazard frequency
- hurricane-induced plant impacts
- calculation of hurricane-induced CDFs and release frequencies

Similar to the full-power analysis, the tornado hazard frequency is based on NOAA data. The high-wind strike frequency does not reflect seasonal or geographic variations. The tornado strike frequencies were estimated by taking the number of occurrences of a tornado for a given enhanced Fujita Scale rating, divided by the square miles of the U.S., divided by the number of years of data, multiplied by the site area. The shutdown tornado frequencies were then estimated by multiplying the strike frequencies by the number of hours per calendar year that the plant is expected to be in each shutdown plant operating state.

As in the full-power analysis, GEH used NUREG/CR-6890 to develop data for loss of offsite power resulting from hurricanes. The initiating event frequency for hurricane-related loss of offsite power events is the number of loss of offsite power events caused by hurricanes at power divided by the reactor years for coastal plants. For each shutdown plant operating state, the full-power hurricane frequency is multiplied by the number of hours per calendar year that the plant is expected to be in each shutdown plant operating state, assuming one outage in every 2-calendar-year period.

GEH used the shutdown PRA accident sequence structures, system fault trees, and success criteria to calculate shutdown high-winds CDF and releases. The analysis included operator actions, such as the operator failing to recognize the need for the non-safety-related low-pressure injection makeup after depressurization. The analysis also explicitly quantified tornado and hurricane scenarios during Mode 5, Mode 5 (Open), and Mode 6 (Unflooded). The accident sequence analysis did not explicitly quantify scenarios during Mode 6 (Flooded) since adequate water will remain above the core to prevent core damage for more than 24 hours.

GEH estimated the mean high-winds shutdown CDF be $1x10^{-9}$/yr. Since the containment was assumed to be open during Modes 5 and 6, this CDF is also the LRF. A hurricane-induced loss of offsite power during Mode 5 and Mode 5 (Open) accounts for over 98 percent of the shutdown high-winds CDF.

GEH did not assess the high-winds risk during Mode 6 (Unflooded) operation. Because of the long advance warning for an approaching hurricane, GEH assumed that the plant staff will have adequate time to, at a minimum, set the head in place, resulting in a Mode 5 (Open) configuration. The model reflects this assumption through the addition of the shutdown hurricane Mode 6 (Unflooded) high-wind strike frequency to that for the shutdown hurricane Mode 5 (Open). GEH documented this assumption as a key risk insight from the analysis.

19.1.6.2.1.5 Shutdown Seismic Significant Accident Sequences and Leading Contributors

Similar to the full-power assessment, GEH performed a shutdown SMA to calculate HCLPF seismic capacities for important accident sequences and accident classes. The PRA-based seismic margins approach used in this analysis evaluates the capability of the plant to withstand an earthquake of 1.67 times the SSE. GEH used the MIN-MAX method to determine the functional and accident sequence fragilities.

The HCLPF nodal fault trees used for the shutdown seismic analysis are the same as those used in the full-power seismic analysis, with the exception of the structural failure node. The accident sequence analysis assumed the earthquake-induced initiating event to be an LOPP. The model assumes that scenarios with structural failures will lead directly to core damage. As in the shutdown internal events assessment, GEH developed shutdown seismic event trees for plant operational states, Mode 5, Mode 5 (Open), Mode 6 (Unflooded), and Mode 6 (Flooded). No shutdown accident sequence has an HCLPF lower than 0.84 g. The staff concurs that GEH has shown the ESBWR plant and equipment to be capable of withstanding an earthquake with a magnitude at least 1.67 times the SSE.

19.1.6.2.2 Risk-Significant Functions/Design Features, Phenomena/Challenges, and Human Actions

19.1.6.2.2.1 Shutdown Fire Assessment

A fire in the MCR will not result in a shutdown initiator. The ESBWR MCR is designed differently from the traditional MCR. The ESBWR MCR controls are connected to the back panel rooms via fiber-optic cables, which are unaffected by an MCR fire. The loss (including melting) of the cables or visual display units will not cause inadvertent actuations or affect the automatic actions associated with safety and non-safety equipment.

To limit spurious actuations of safety-related equipment, the hard wires are minimized to control the consequences of a postulated fire. From the DCIS rooms to the components, fiber optics will also be used up to the RMUs in the plant. Hard wires are then used to control the subject components. Typically, two load drivers are actuated simultaneously to actuate the component. To eliminate spurious actuations, these two load drivers are located in different fire areas. Therefore, by design, a fire in a single fire area cannot cause spurious actuation of safety-related equipment.

Regarding the treatment of fires in primary containment during shutdown, the small quantity of combustible materials and spatial separation is assumed to prevent damage to the redundant divisional circuits in this area. During shutdown, the primary containment is deinerted. The Level 2 PRA considers deinerted operation before and following shutdown as described in Section 8.1.4 of the PRA report.

Regarding fires in the drywell/containment area, this area was screened from the shutdown fire assessment. GEH assumed that a fire in the drywell/containment area is highly unlikely to result in the loss of RWCU/SDC. The RWCU system inboard containment isolation valves are located in the LDW, which could be well separated spatially according to GEH. GEH also believes that minimal combustible fuel loads will be located inside the LDW. Screening of a postulated drywell/containment fire that could result in a loss of RWCU/SDC and the RWCU inboard containment isolation valves is risk significant. GEH did not identify spatial separation of the RWCU containment isolation or limiting combustible loading in the drywell containment area as a key risk insight in Table 19.2-3 of DCD Tier 2, Revision 4. In RAI 19.1-4, Supplement 1, the staff requested GEH to address this issue. **RAI 19.1-4, Supplement 1 is being tracked as an open item.**

Based on staff review of the risk achievement results, the shutdown FPRA results are not as sensitive to operator errors as CCFs in the GDCS, which have the highest RAWs.

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.1.6.2.2.2 Shutdown Floods Assessment

GEH assumed that floods in containment during shutdown will not affect ICS components or the DPVs because these components are relatively high in the containment. As a result, the staff found that the top 200 shutdown flooding cutsets occur during Mode 6 (Unflooded) when the ICS is not available for DHR. Based on staff review of the risk achievement results, the shutdown flood PRA results are not as sensitive to operator errors as CCFs in the GDCS, which have the highest RAWs.

19.1.6.2.2.3 Shutdown High-Winds Assessment

As discussed in Section 19.1.6.2.1.4, of this report, a hurricane-induced loss of offsite power during Mode 5 and Mode 5 (Open) accounts for more than 98 percent of the shutdown high-winds CDF. GEH assumed that during Mode 6 (Unflooded) operation, the plant staff will have adequate time to, at a minimum, set the head in place, resulting in a Mode 5 (Open) configuration. Based on staff review of the risk achievement results, the shutdown high-winds PRA results are not as sensitive to operator errors as CCFs in the GDCS and the uninterruptible ac power supply, which have the highest RAWs.

19.1.6.2.3 Conclusions

Based on the discussions above, the staff concludes that the methodology and approach of the external events low-power and shutdown risk analysis are acceptable Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability of identification of design insights.

19.1.7 Probabilistic Risk Assessment-Related Input to Other Programs and Processes

19.1.7.1 <u>Probabilistic Risk Assessment Input to the Design Certification Process</u>

19.1.7.1.1 Summary of Technical Information

The applicant used the PRA insights and assumptions to develop a list of DC requirements. The DCD incorporates these requirements in Table 19.2, as appropriate, to ensure that any future plant that references the ESBWR design will be built and operated in a manner consistent with the important assumptions made in the ESBWR DC PRA.

19.1.7.1.2 Regulatory Criteria

The staff evaluated the PRA input to the DC process against the Commission’s objectives for new reactor designs stated in Section 19.1.1 of this report. The following three objectives are especially relevant:

(1) Develop an in-depth understanding of design robustness and tolerance of severe accidents initiated by either internal or external events.

(2) Develop a good appreciation of the risk significance of human errors associated with the design and characterize the key errors in preparation for better training and more refined procedures.

(3) Identify important safety insights related to design features and assumptions made in the PRA to support certification requirements, such as ITAAC, Design Reliability Assurance Program (D-RAP) requirements, and TSs, as well as COL and interface requirements.

19.1.7.1.3 Staff Evaluation

The applicant achieved the first two objectives by identifying the dominant accident sequences, as well as the risk-important design features and human actions (see Sections 19.1.3 to 19.1.5 of this report).

The staff reviewed the list of DC requirements and determined that it does not reflect all of the important assumptions made in the PRA. The staff issued an RAI to the applicant in order to understand why certain assumptions and insights were not translated into DC requirements. **This issue is being tracked as Open Item 19.1.7.1-1.**

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.1.7.2 <u>Probabilistic Risk Assessment Input to the Maintenance Rule Implementation</u>

Importance measures are derived from the PRA and used to develop a list of risk-significant SSCs for the ESBWR DC, as discussed in Section 17.4.6 of the DCD. Section 17.4 of this report documents the staff's evaluation of the information provided in Section 17.4.6 of the DCD.

19.1.7.3 <u>Probabilistic Risk Assessment Input to the Reliability Assurance Program</u>

The ESBWR D-RAP is a program utilized during detailed design and specific equipment selection phases to ensure that the important ESBWR reliability assumptions of the PRA are considered throughout the plant life. The PRA is used to evaluate the plant response to anticipated operational occurrence initiating events and mitigation to ensure that potential plant damage scenarios pose a very low risk to the public. The D-RAP identifies relevant aspects of plant operation, maintenance, and performance monitoring of important plant SSCs for owner/operator consideration in assuring safety of the equipment and limiting risk to the public. Importance measures derived from the PRA are used to develop a list of risk-significant SSCs for the ESBWR DC, as discussed in Section 17.4.6 of the DCD. Section 17.4 of this report documents the staff's evaluation of the D-RAP and the applicant's use of the PRA to support the program.

19.1.7.4 Probabilistic Risk Assessment Input to the Regulatory Treatment of Non-safety-Related Systems Program

The ESBWR design process uses a systematic approach to identify regulatory guidance and assess it relative to specified ESBWR design features to determine whether additional regulatory treatment is warranted for SSCs that perform a significant safety, special event, or post-accident recovery function. The ESBWR design process includes the use of both probabilistic and deterministic criteria to achieve the objectives of SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs”, dated March 28, 1994. Criterion C requires an assessment of safety functions that are relied upon at-power and during shutdown conditions to meet the NRC's safety goal guidelines. A comprehensive assessment to identify RTNSS candidates includes focused PRA sensitivity studies for internal events, evaluations of external events, an assessment of the effects of non-safety-related systems on initiating event frequencies, and an assessment of uncertainties in these analyses and uncertainties that may be introduced by first-of-a-kind passive components. Chapter 22 of this SER documents the staff’s evaluation of the focused PRA studies used to support the RTNSS process.

19.1.8 Conclusions

The NRC has evaluated the ESBWR PRA and its use in the design and certification processes. The staff finds that, while the applicant has addressed the Commission’s objectives described in Section 19.1.1 of this report, a number of open issues must be resolved before the staff can finalize its review. The staff has described each open item in the appropriate section of the report. Because of the open items that remain to be resolved, the staff is unable to finalize its conclusions regarding acceptability of the applicant’s PRA and its use in the design and certification processes.

19.2 Severe Accident Evaluations

19.2.1 Regulatory Criteria

The staff reviewed the applicant’s description and analysis of the design features to prevent and mitigate severe accidents, in accordance with the requirements in paragraph 23 of 10 CFR 52.47, “Contents of Applications; Technical Information.” This review covered specific issues identified in SECY-90-016 and SECY-93-087, which the Commission approved in related SRMs dated June 26, 1990, and July 21, 1993, respectively, for prevention (e.g., ATWS, midloop operation, SBO, fire protection, and interfacing system LOCA (ISLOCA)) and mitigation (e.g., hydrogen generation and control, core debris coolability, high-pressure core melt ejection, containment performance, dedicated containment vent penetration, and equipment survivability).

In addition, the staff reviewed the information the applicant provided to satisfy the requirements of paragraph 8 in 10 CFR 52.47.

19.2.2 Severe Accident Prevention

19.2.2.1 Severe Accident Prevention Features

Section 19.1.3.1 of this report summarizes important features.

19.2.2.1.1 Anticipated Transients Without Scram

For ATWS prevention and mitigation, the ESBWR is designed with the following features:

- an ARI system that utilizes sensors and logic that are diverse and independent of the RPS
- electrical insertion of FMCRDs that also utilize sensors and logic that are diverse and independent of the RPS
- automatic feedwater runback under conditions indicative of an ATWS
- automatic initiation of SLC under conditions indicative of an ATWS
- elimination of the scram discharge volume in the CRD system

DCD Section 15.5.4 discusses the effectiveness of these design features for addressing ATWS concerns. Given these features, the ESBWR PRA demonstrates that ATWS provides an insignificant contribution to CDF and LRF.

19.2.2.1.2 Midloop Operations

Not applicable.

19.2.2.1.3 Station Blackout

During a total loss of offsite power, the safety-related electrical distribution system is automatically powered from the onsite non-safety-related diesel generators. If these diesel generators are not available, then each division of the safety-related system independently isolates itself from the non-safety-related system, and the safety-related batteries of each division provide uninterrupted power to safety-related loads of each safety-related load division. The divisional batteries are sized to provide power to required loads for 72 hours. DCD Section 15.5.5 documents conformance to the requirements of 10 CFR 50.63, “Loss of All Alternating Current Power.” Because of the nature of the passive safety-related systems in the ESBWR, SBO events are not significant contributors to CDF or LRF.

19.2.2.1.4 Fire Protection

The FPS does not perform any safety-related function. The FPS serves as a preventive feature for severe accidents in two ways: (1) by reducing or eliminating the possibility of fire events that could induce transients, damage mitigation equipment, and hamper operator responses, and (2) as a means for long-term makeup to the upper containment pools, which may be required after the first 72 hours of an accident requiring passive heat removal.

The FPS connects to the safety-related portion of the FAPCS. The FPS has RTNSS functions that provide post 72-hour makeup to the IC/PCC pools and spent fuel pool using this portion of the FAPCS. The FPS primary water storage tank also has the RTNSS function of providing makeup water for reactor coolant inventory.

Section 19.1.5 of this report summarizes the risk significance of fire. Performance of RTNSS functions, and the piping supporting these functions, is assured by applying the augmented design standards (Category B1) described in DCD Section 19A.8.3.

19.2.2.1.5 Intersystem Loss-of-Coolant Accident

As stated earlier in Section 19.1.3.1, the design of the ESBWR reduces the possibility of ISLOCA outside containment by designing all piping systems, major system components, and subsystems connected to the RCPB to have ultimate rupture strength at least equal to the RCPB pressure. Given these design features, ISLOCA is not a significant contributor to initiating events or accidents.

19.2.2.1.6 Alternating Current-Independent Fire Water Addition System

The FPS not only plays an important role in preventing core damage, but it is also the backup source of water for flooding the LDW should the core become damaged and relocate into the containment (the primary source is the deluge subsystem pipes of the GDCS). The primary injection path is through the feedwater line and into the RPV. This system must be manually aligned. This is appropriate because the sequences in which it is useful are slow to develop and easy to identify.

19.2.2.1.7 Vessel Depressurization

Section 19.1.3.1 describes this issue.

19.2.2.1.8 Isolation Condenser

Section 19.1.3.1 describes this issue.

19.2.3 Severe Accident Mitigation

19.2.3.1 Overview of Containment Design

Figure 19.2-1 illustrates the ESBWR containment design features that would mitigate severe accidents, and Section 19.1.3.1.1 discusses the major features.

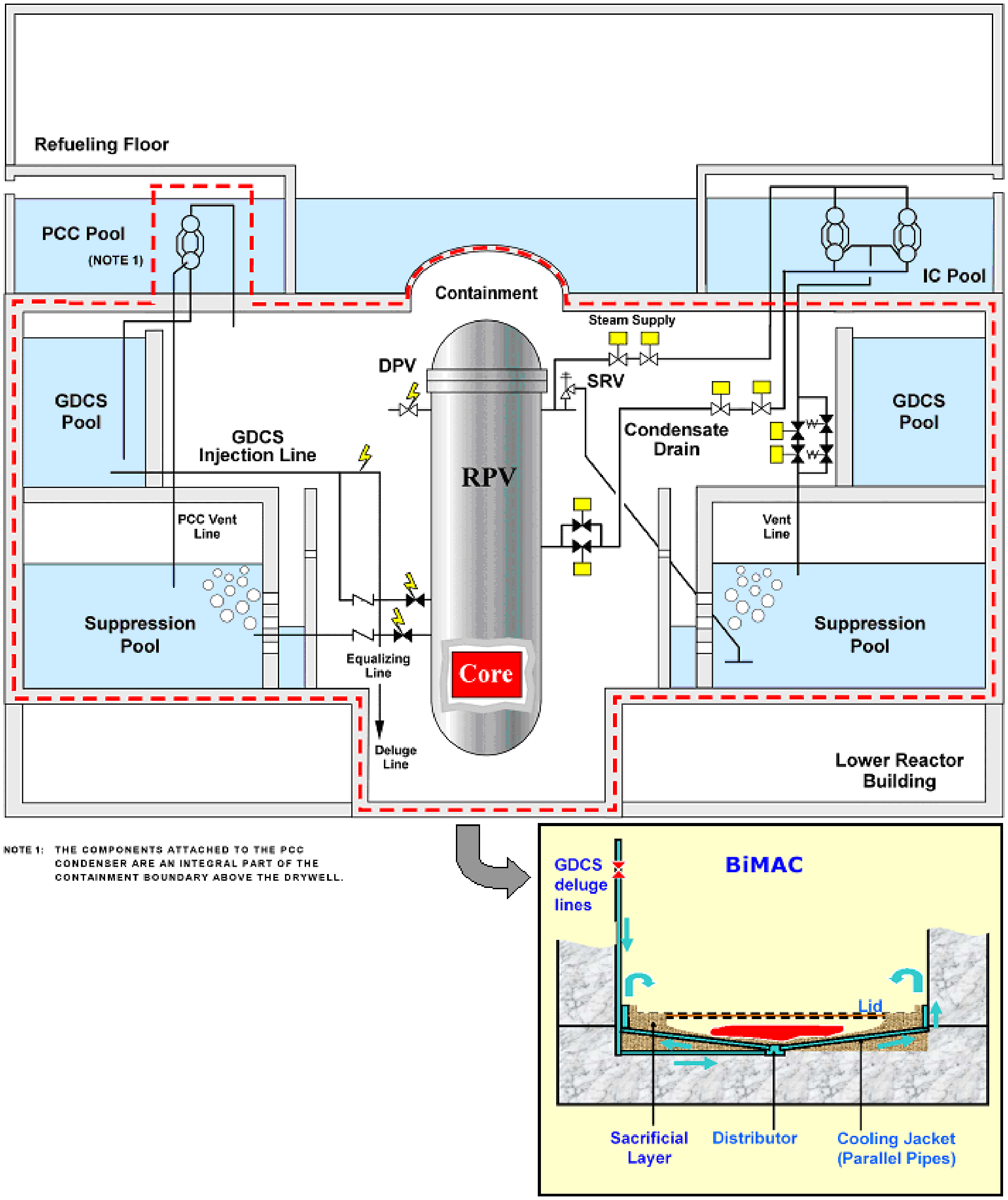

Figure 19.2-1 ESBWR design features for severe accident conditions

19.2.3.2 <u>Severe Accident Progression</u>

Severe accident progression can be divided into two phases: an in-vessel stage and an ex-vessel stage. The in-vessel stage generally begins with insufficient DHR and can lead to melt-through of the reactor vessel. The ex-vessel stage involves the release of the core debris from the reactor vessel into the containment and resulting phenomena such as CCI, FCI, and DCH.

19.2.3.2.1 In-Vessel Melt Progression

In-vessel melt progression establishes the initial conditions for assessing the thermal and mechanical loads that may ultimately threaten the integrity of the containment. In-vessel melt progression begins with uncovering of the core and initial heatup and continues until either (1) the degraded core is stabilized and cooled within the reactor vessel, or (2) the reactor vessel is breached and molten core material is released into the containment. The phenomena and processes in the ESBWR that can occur during in-vessel melt progression include the following:

- core heatup resulting from loss of adequate cooling
- exothermic metal-water reactions that oxidize cladding and produce hydrogen
- eutectic interactions (i.e., mixtures of materials with a melting point lower than that of any other combination of the same components) between core materials (e.g., control blades and fuel assembly channel boxes, resulting in relocation of molten material)
- melting and relocation of cladding, structural materials, and fuel
- formation of blockages near the bottom of the core resulting from the solidification of relocating molten materials
- drainage of molten materials to the vessel lower head region
- formation of a melt pool, natural circulation heat transfer, crust formation, and crust failure in the lower head region
- lower head breach resulting from failure of a penetration or from local or global creep-rupture

As the temperature of the core increases, fission products in vapor form are released. As the vapors rise, they condense into liquid aerosols, which can either be deposited on surfaces, such as upper internal structures, or flow along with the steam and hydrogen out of the RPV, either through the SRV lines to the suppression pool during RCS depressurization or through breaks in the RCS boundary.

The core melt progression, including relocation and fission product release, becomes increasingly difficult to predict as the core continues to degrade. The core melt could relocate into the lower reactor vessel plenum. If water is present in the lower plenum, the potential exists for in-vessel steam explosions, where molten fuel rapidly fragments and transfers its energy, causing rapid steam generation and shock waves. Another possibility is that the core debris within the lower plenum may melt through the reactor vessel or interact with available water before melting through and entering the LDW.

19.2.3.2.2 Ex-Vessel Melt Progression

Ex-vessel severe accident progression is affected by the mode and timing of the reactor vessel failure; the primary system pressure at reactor vessel failure; the composition, amount, and character of the molten core debris expelled; the type of concrete used in containment construction; and the availability of water to the LDW. The initial response of the containment to ex-vessel severe accident progression is largely a function of the pressure of the RCS at reactor vessel failure and the existence of water within the reactor cavity. If not prevented through design features, risk consequences are usually dominated by early CF mechanisms that could result from energetic severe accident phenomena such as high-pressure melt ejection (HPME) with DCH and EVEs. The long-term response of the containment from ex-vessel severe accident progression is largely a function of the containment pressure and temperature resulting from CCI and the availability of CHR mechanisms.

At high RCS pressures, the molten core debris could be ejected from the reactor vessel in jet form causing it to fragment into small particles. The potential exists for the core debris ejected from the vessel to be swept out of the LDW and into the UDW. Finely fragmented and dispersed core debris could heat the containment atmosphere and lead to large pressure spikes. In addition, chemical reactions of the core debris particulate with oxygen and steam could add to the pressurization loads. This severe accident phenomenon is known as HPME with DCH.

To prevent this phenomenon, the ESBWR has incorporated a reliable depressurization system to provide assurance that, in the event of a core melt scenario, failure of the RPV would occur at a low pressure. Should the RPV fail at a high pressure, the design of the ESBWR containment would provide an indirect pathway from the LDW to the UDW in an effort to decrease the amount of core debris that could contribute to DCH.

RPV failure at high or low pressure coincident with water present within the LDW could lead to FCI with the potential for rapid steam generation or steam explosions. Rapid steam generation involves the pressurization of containment compartments from nonexplosive steam generation beyond the capability of the compartment to relieve the pressure so that local overpressurization failure of the compartment occurs. Steam explosions involve the rapid mixing of finely fragmented core debris with surrounding water, resulting in rapid vaporization and acceleration of the surrounding water creating substantial pressure and impact loads. The ESBWR is designed so that there is a very low likelihood of water within the LDW at the time of reactor vessel failure.

The ESBWR has incorporated a passive debris cooling device, the BiMAC, to cool debris once it enters the LDW (see Section 19.2.3.3.3). Without such a device, contact of molten core debris with concrete in the LDW would lead to CCI. CCI involves the decomposition of concrete from core debris and can challenge the containment though various mechanisms, including (1) pressurization resulting from the production of steam and noncondensable gases to the point of containment rupture, (2) the transport of high-temperature gases and aerosols into the UDW leading to high-temperature failure of the containment seals and penetrations, (3) liner melt-through, (4) reactor pedestal melt-through leading to relocation of the reactor vessel and tearing of containment penetrations, and (5) the production of combustible gases such as hydrogen and carbon monoxide. CCI is affected by many factors, including the availability of water to the LDW, the containment geometry, the composition and amount of core melt, the core melt superheat, and the type of concrete involved.

19.2.3.3 Severe Accident Mitigative Features

The ESBWR containment has been designed with specific mitigating capabilities. These capabilities not only mitigate the consequences of a severe accident but also address uncertainties in severe accident phenomena. Section 19.1.3 of this report describes these features and discusses the specific severe accident phenomena addressed by the mitigation system.

The following discussion is an evaluation of how the ESBWR design addresses the severe accident mitigative features issues, including those raised in SECY-90-016 and SECY-93-087.

19.2.3.3.1 External Reactor Vessel Cooling

As one potential option for arresting the melt propagation process and ensuring long-term coolability within the containment boundary, the applicant examined the applicability and effectiveness of in-vessel retention already developed and utilized for the passive PWR designs in the U.S. GEH concluded that this could be a highly effective approach for the ESBWR as well. However, this approach would require all equipment found hanging from the lower head penetrations to be supported from the outside so as to maintain the melt-containing capacity of the lower head. This proved unworkable from an operational perspective, so the option was rejected.

19.2.3.3.2 Hydrogen Generation and Control

The analysis of the radiolytic oxygen concentration in containment, as discussed in Section 8.1 of the PRA report, Revision 2, is based on the methodology of Appendix A to SRP Section 6.2.5 and RG 1.7, “Control of Combustible Gas Concentrations in Containment.”

The analysis results show that the time required for the oxygen concentration to increase to the deinerting value of 5 percent is significantly greater than 24 hours for a wide range of fuel cladding-steam interaction and iodine release assumptions of up to 100 percent of the initial core inventory.

Therefore, the Level 2 PRA does not take credit for venting to prevent unacceptable hydrogen and oxygen concentrations in the drywell or the suppression chamber. Venting for pressure relief is modeled as an operator action (i.e., no mechanical faults).

19.2.3.3.2.1 Preventive and/or Mitigative Features

In the ESBWR, the containment inerting system is provided to establish and maintain an inert atmosphere within the containment. This inerting prevents the combustion of hydrogen. The containment is inerted during operation, except for short periods immediately before and after scheduled shutdowns when the containment is deinerted to establish a clean, breathable atmosphere throughout the containment while the containment is still closed.

19.2.3.3.2.2 Risk Caused by Deinerted Operation

The PRA analysis assumes a 24-hour-per-year period of noninerted containment atmosphere. This adds an additional BYP frequency of $3.3x10^{-11}$/yr.

19.2.3.3.2.3 Basis for Acceptability

The specific requirements in 10 CFR 50.44(c)(2) establish the following for future water-cooled reactor applicants and licensees:

> all containments must have an inerted atmosphere, or must limit hydrogen concentrations in containment during and following an accident that releases an

equivalent amount of hydrogen as would be generated from a 100 percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) and maintain containment structural integrity and appropriate accident mitigating features.

The design of the ESBWR provides for inerted containment and, as a result, requires no system to limit hydrogen concentration.

The ESBWR containment, per 10 CFR 50.34(f)(2)(ix), can withstand the pressure and energy addition during and following an accident that releases an amount of hydrogen equivalent to that generated from a 100-percent fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume) and maintain containment structural integrity and appropriate accident mitigating features.

In SECY-00-0198, "Status Report on Study of Risk-Informed Changes to the Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-informed Changes to 10 CFR 50.44 (Combustible Gas Control)," dated September 14, 2000, the NRC staff recommended changes to 10 CFR 50.44, "Combustible Gas Control for Nuclear Power Plants," to reflect the position that only combustible gas generated by a beyond-design-basis accident is a risk-significant threat to containment integrity.

During severe accident conditions with a significant amount of fission product gases and hydrogen release to the containment, the containment will remain inerted without any additional action because radiolytic oxygen production remains below the concentration that could pose a risk of hydrogen burning for a significant period of time following the event. Accumulation of combustible gases that may develop in the period after about 24 hours will be managed by implementation of the severe accident management guidelines. For a severe accident with a substantial release of hydrogen, the oxygen concentration in containment from radiolysis is not expected to reach 5 percent for significantly longer than 24 hours.

According to 10 CFR 50.44(c)(2), which provides the combustible gas control requirements for future water-cooled reactor applicants and licensees, containments with an inerted atmosphere do not require a method to control the potential buildup of postaccident hydrogen.

The ESBWR PRA for severe accidents considers gas generation effects, combustible and noncombustible commingled, for situations in which they can possibly lead to overpressure by their molar additions to the containment atmosphere. The calculated frequency of such failures is acceptably small, as noted in Section 19.1.4.2 of this report.

The present review concurs that, for ESBWR operations at power with the containment inerted, combustion of hydrogen and other combustible gases does not have to be considered as a safety risk. The ESBWR design is in compliance with the Commission's safety goals and regulations regarding hydrogen combustion and control.

Risk and safety of operations when the containment is not inerted (e.g., when the containment is open during shutdown) will need to be considered because combustible gas generation can occur in locations with oxygen present at combustion-supporting levels.

19.2.3.3.3 Core Debris Coolability

In severe accidents that proceed to vessel failure and release molten core material into the containment, the in-vessel melt progression establishes the initial conditions for assessment of the thermal and mechanical loads that may ultimately challenge the integrity of the containment. The end stages of the in-vessel process are the formation of a melt pool in the vessel lower head region, subsequent lower head breach resulting from failure of a penetration or from local or global creep-rupture, and relocation of the molten material into the LDW region. The initial response of the containment to ex-vessel severe accident progression is largely a function of the pressure of the RCS at reactor vessel failure and the existence of water within the reactor cavity.

For all currently operated LWRs, the severe accident management case is based on the basic premise that, provided a sufficient floor area available for spreading and a sufficient amount of water to cover the molten core debris, the debris will become quenched and will remain coolable thereafter. While ESBWR satisfies the basic conditions for this approach (i.e., the core melt spreadable floor area according to the EPRI URD guidelines for advanced reactors), the core-on-the-floor approach is further improved. GEH has incorporated design features (e.g., the BiMAC device) that, according to the applicant, make the issue of corium-concrete interactions, along with the great uncertainties that arise in its consideration, inconsequential.

19.2.3.3.3.1 Basemat Internal Melt Arrest and Coolability Device Design

The ESBWR design uses a passively cooled boundary that is designed to be impenetrable by the core debris on the LDW floor. This device is called the BiMAC. The boundary is made by a series of inclined pipes, placed side by side, forming a jacket that can be effectively and passively cooled by natural circulation when subjected to thermal loading on any portion(s) of it. Water is supplied to this device from the GDCS pools via a set of squib-valve-activated deluge lines. The timing and flows are such that (1) cooling becomes available immediately upon actuation, and (2) the chance of flooding the LDW prematurely, to the extent that such an event results in a vulnerability to steam explosions, is very remote. The jacket is buried inside the concrete basemat and would be called into action only in the event that some or all of the core debris on top is noncoolable.

The paragraphs below describe important considerations in the implementation of this concept.

Pipe inclination angle. Both the thermal load caused by melt natural circulation and the burnout CHF, increase with an angle of inclination θ of the bottom boundary from the very low values pertinent for a perfectly horizontal orientation. This increase is much faster for the CHF in the region $0<\theta<20°$, and there is a maximum separation around the upper end of this range. Within a reasonable value of the overall vertical dimension of the BiMAC device, the whole LDW can be covered conveniently with pipes inclined at near the upper end of this range.

Sacrificial refractory layer. A refractory material is laid on top of the BiMAC pipes to protect against melt impingement during the initial (main) relocation event and to allow some adequately short time for diagnosing that conditions are appropriate for flooding. This approach will minimize the chance of inadvertent, early flooding. The selected material, such as ceramic zirconia, will have high structural integrity and high resistance to melting.

Melt jet impingement. Heat transfer and related phase change processes during melt jet impingement on a solid slab have been studied in the past and their mechanisms are well understood. Notably, because of the high melting point of the jet's liquid, compared to the slab's initial temperature, a crust is formed and serves as a thermal boundary condition through which the heat transfer occurs. As stated above, BiMAC is protected by a high-temperature melting point refractory material cover to eliminate any challenges resulting from impingement of the superheated, metallic melt jets on the BiMAC cooling pipes. As opposed to being ablated and swept away by the jet flow, the refractory layer will be designed to remain structurally intact and to limit the heat transfer by conduction in a low-conductivity medium.

The BiMAC cavity. The coolable volume, up to the height of the vertical segments of the BiMAC pipes, is approximately 400 percent of the full-core debris. Thus, no possibility exists for the melt to contact the LDW liner; melt can only go into the BiMAC. There is complete floor coverage.

Sump protection. GEH stated that the two sumps needed for detecting leakage flow during normal operation are positioned and protected, as is the rest of the LDW liner, from being subject to melt attack. There are two sumps, shaped and positioned next to the pedestal wall so that they offer no significant “target” to the melt stream exiting the vessel under most release scenarios.

The LDW deluge system. This system consists of three main lines that feed off the three independent GDCS pools, respectively, each separating into a pair of lines. One from each pair of these lines connects to the BiMAC main header, the other discharges directly into the LDW from near the top. There are six valves, one for each line. After receiving signals from numerous thermocouples/conductivity probes that cover the LDW floor area and air space, three of the valves (i.e., the valves on lines that feed into the BiMAC) are operated to a sufficient degree to indicate melt arrival following RPV breach. In the event of a vessel breach away from the very bottom of the lower head, the quantity of melt, the driving force (low-pressure scenario), and the chance of direct impact would be small and thus insufficient to damage the deluge pipes. The valves on lines that feed directly into the LDW will be designed to operate on a diverse detection and activation system. These lines are sized so that any three of them would be sufficient to ensure proper BiMAC functioning (i.e., operation in the natural circulation mode within 5 minutes from melt arrival on the floor). The required reliability of the system (at a high confidence level) is that its failure on demand is not to exceed $1x10^{-3}$. The detection and actuation of the system is to be completed at the COL stage of the design.

Successful functioning of the BiMAC device depends on the condition that heat removal capability by boiling exceeds the thermal loading resulting from melt natural convection. In addition, it must be shown, through test or analyses, that at the end of the main melt relocation event and associated ablation process the BiMAC sacrificial layer is left with some material still protecting the steel pipes.

The BiMAC concept is based on sound analytical considerations built on top of separate-effects experiences on burnout heat fluxes in inverted geometries and two-phase (boiling) pressure drop in inclined pipes. Nevertheless, the limits of coolability are defined by the burnout heat flux, or CHF, of water boiling on the inside of the inclined BiMAC pipes. The CHF increases rapidly with angle of inclination, and this increase is most rapid in the 0-to-20-degrees interval.

The applicant carried out a testing program to demonstrate that the BiMAC device would effectively remove the decay heat in the core debris and thus confirm the design. The staff has requested documentation of the test results in RAIs 19.2-23 S02 and 19.2-25 S02. **RAIs 19.2-23 S02 and 19.2-25 S02 are being tracked as open items.**

19.2.3.3.3.2 Conclusion

The ESBWR PRA report describes the detailed probabilistic framework, quantification of basemat penetration loads, quantification of fragility to BMP, and prediction of failure probability caused by BMP. The results of the BMP device analysis described in NEDO-33201 show that the BiMAC device would be effective in containing all core melts in a manner that ensures long-term coolability and stabilization of the resulting debris. In this way, the concrete basemat penetration issue becomes moot, as is containment overpressurization by the so-generated concrete decomposition gases. However, due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.2.3.3.4 High-Pressure Melt Ejection

At high RCS pressures at the time of RPV failure, a potential exists for the core debris ejected from the vessel to be swept out of the LDW and into the UDW. Finely fragmented and dispersed core debris could heat the containment atmosphere and lead to large pressure spikes. In addition, chemical reactions of the core debris particulate with oxygen and steam could add to the pressurization loads. This severe accident phenomenon is known as HPME with DCH.

In the ESBWR, the UDW is vented to another volume, the wetwell, which contains a large and effective heat sink. As the ESBWR is inerted, any combustion of hydrogen and resulting pressurization loadings is limited to the amount of residual oxygen present within the containment atmosphere.

No specific ESBWR containment design features address the DCH loads other than the general arrangement of the drywell, wetwell, and connecting vents paths.

The set of potential accidents that lead to DCH consists of those involving core degradation and vessel failure at high primary system pressure (the Class III scenarios). The probability of the necessary preceding combinations of events is assessed through the ROAAM process as remote and speculative; that is, they could, without further ado, be left in the category of residual risks. Still, because of its potentially severe consequences, the applicant chose to further examine the likelihood for energetic CF from DCH, and concluded, by analysis, that such a failure is PU.

The key ingredient towards such a conclusion is that approximately 14 square meters (m^2) of vent area, connecting to the condensation potential of the suppression pool, makes it virtually impossible to overpressurize the drywell volume. Just as in a LOCA, the timing of “vent clearing” is important.

The applicant also examined the potential for liner failure resulting from the associated high temperatures in the drywell. For the UDW liner, this type of failure was also found to be PU, while for the LDW, because of the immediate proximity and contact with large quantities of melt (given an HPME), local failures, although highly unlikely, cannot be excluded. The consequences of such a possibility is limited by a standard design feature (anchoring) which compartmentalizes the liner and isolates the gap space of the LDW from that of the UDW, clearly eliminating any flowpaths to the outside.

The applicant adapted an existing analytical model to establish the transient containment conditions. The model equations are simple mass and energy balances over the communicating LDW, UDW, and wetwell volumes. This model is verified by comparison with final pressures/temperatures calculated for the original closed system configuration of the original model, as well as sample test results.

Ablation of the initial penetration opening (and of the sacrificial refractory layer on top of the BiMAC during HPME) is estimated according to established models and procedures. The results for vessel hole ablation are very similar to those obtained previously, yielding final diameters of 0.2 meter and 0.3 meter for 100 and 300 tons of melt involved in the expulsion process, respectively. These results established the rates of the driving steam escape from the vessel. The containment-limiting fragility is failure of the drywell head.

The margin to failure is the difference between the bounding estimates of loads (upper bound) and fragility (lower bound). The results show that overpressure (catastrophic) failure of the ESBWR containment from DCH is PU in terms used in the ROAAM process. This conclusion covers all Class III accidents.

During normal operation, the UDW head is immersed in a water pool, and it remains cold during the duration of a high-pressure meltdown sequence. Bounding estimates of this process yield internal DW head temperatures of less than 450 K. Thermally induced failure of the UDW head and/or its seals is thus also PU for all Class III accidents.

Thermally induced failure of the liner, including the penetration areas, is relevant to Class III accidents in which drywell spray is assumed to be unavailable, and these sequences amount to approximately 1 percent of the CDF. GEH finds that, even in these cases, strains caused by thermal stress are rather modest (less than 8 percent) in relation to what might be considered necessary for cracking or tearing, even at temperatures approaching the melting point of the material. Bounding calculations of DCH-induced UDW temperatures indicate that the relevant temperature levels are -1000 K, which is considerably below the near-melting temperatures (over 1650 K) that could cause failure.

However, the GEH calculations also show short periods of potentially very high temperatures in the LDW atmosphere (up to 4000 K). These temperatures, and the presence of potentially large quantities of melt in the LDW, indicate that the LDW liner could be subject to local failures, a condition that is noted in the high-pressure CET. The branch is only used in a Level 3 sensitivity study.

The staff accepts that the exclusion of DCH-induced catastrophic CFs is reasonable. The staff also agrees that a high probability of localized liner failures in the LDW exists.

19.2.3.3.5 Fuel Coolant Interactions

The containment function may be challenged by a rapid energy release during a FCI that results in a steam explosion. The term "steam explosion" refers to a phenomenon in which molten fuel rapidly fragments and transfers its energy to the coolant, resulting in rapid steam generation, shock waves, and possible mechanical damage. To be a significant safety concern, the interaction must be very rapid and must involve a large fraction of the core mass. Steam explosions may occur either in the vessel or outside the vessel.

19.2.3.3.5.1 In-Vessel Steam Explosion

The in-vessel steam explosion is essentially of exclusive interest to PWRs. The Steam Explosions Review Group (SERG) convened by the NRC in 1985 as SERG-1, and again in 1995 as SERG-2, focused on the alpha-mode CF (α-failure). The SERG only considered the issue of in-vessel steam explosions for PWRs in detail. For BWRs, the lower plenum design, largely and densely occupied by control rod guide tubes, is considered to be generically prohibitive of the large-scale events required for α-failure. This conclusion is also applicable to the ESBWR design.

19.2.3.3.5.2 Ex-Vessel Steam Explosion Effects

EVEs are energetic FCIs that are triggered from already premixed states developed as the melt released from the RPV falls into and traverses the depth of a water pool below. In BWRs, LDW designs have traditionally employed very large-height geometries, which, when flooded, form deep water pools below the reactor vessel. Furthermore, metallic melts, such as those expected in the ESBWR for low-pressure scenarios, are especially prone to energetic interactions. The result is pressure pulses that may reach the kilo-bar magnitude range, potentially capable of loading major structures to failure when large quantities of melt are involved together with highly subcooled water.

Regarding the potential damage from EVE, the relevant structures are the reactor pedestal reinforced concrete wall and the BiMAC device.

Failure of the reactor pedestal, along with the steel liner on it, would constitute violation of the containment boundary. While at static condition, the load-bearing capacity of this structure is adequate; explosive-level pressures acting on millisecond time scales can produce sufficient concrete cracking, along with liner stretching and tearing, to compromise the leak tightness of the containment.

Failure of the BiMAC device on the other hand is defined as crushing of the pipes so that they cannot perform their heat removal function. Such failure would raise the possibility of continuing corium-concrete interactions, basemat penetration, and containment pressurization by the so-generated noncondensable gases.

GEH calculated the fragility of the pedestal under impulse loading using the DYNA3D model, which has been verified and validated for problems of this type. The calculated strains show that, at an impulse load of 600 kPa-s, there is incipient liner failure and noticeable concrete damage. For impulse loadings of 200 and 300 kPa-s, the pedestal holds up well.

GEH carried out calculations for the BiMAC device with the same type of impulse loadings as those used for the reactor pedestal. At impulse loads around 200 kPa-s, a thin portion of the BiMAC embedded pipes yields significantly; however, the remaining material remains basically intact, while the pipe cross-sectional area is still largely intact. This is considered as the level of incipient failure by crushing.

The applicant calculated ESBWR steam explosion impulse rates using the PM-ALPHA.L-3D and ESPROSE.m codes for water pool depths of 1, 2 and 5 meters with 100 K subcooling. With one exception, typical primary impulses on the bottom were approximately 100 kPa-s, while on the side, the impulse magnitudes increase with pool depth from 40 to 150 kPa-s. The loads from 1- and 2-meter deep, highly subcooled pools are taken to bound loads from shallow, saturated pools.

Only the low-pressure-at-vessel breach Class I and Class IV severe accidents have the potential for EVEs. Given the margin between the calculated applied impulses and the structural fragility of the pedestal, GEH concluded through the ROAAM process that pedestal failure by an EVE is PU for pools less than 1.5 meters deep. For accidents with deep (H greater than 1.5 meters), subcooled water pools, GEH stated that an appropriately conservative position would be one in which "integrity of both the liner and the concrete structure could be possibly compromised." In the PRA, this translates to CF for deep pool Class I and Class IV accidents.

The NRC performed independent calculations using the PM-ALPHA/ESPROSE.m computer code to assess the energetics of EVEs for the ESBWR (ERI/NRC 06-202). Fragilities were not recalculated. A base case and four sensitivity cases (assessing different pool depths, vessel breach diameter, and core melt composition) were performed. These calculations produced values of wall (i.e., pedestal) impulse loads ranging from 4 to 60 kPa-s. These values are clearly consistent with and support the GEH estimate of a large margin to CF from EVE for 99 percent of the Class I severe accidents. The basemat (i.e., BiMAC) impulse load was independently calculated to be 35 kPa-s for low pool depths. This is consistent with the GEH "negligible energetics" value and supports the PRA assertion that BiMAC failure is considered PU for low-pressure core melt drops in pools less than 1.5 meters deep.

19.2.3.3.5.3 Minimization of Ex-Vessel Steam Explosion Effects in the ESBWR

The principal element of the GEH ESBWR severe accident management approach on EVE is to minimize the likelihood of deep subcooled water pools in the LDW at the time of vessel failure, including inadvertent spray operation, and to have a structural design capable of coping with the loads expected in cases in which moderate amounts of water (shallow, saturated pools) cannot be avoided.

Containment design prevents subcooled water, to the extent possible, from entering the LDW through the UDW, in particular, by the rerouting of GDCS overflow and by outfitting the wetwell spill-over lines with squib valves, similarly to those that activate the equalizer line. The BiMAC device activation system requires high-temperature thermocouples to detect core-melt arrival and to send signals to actuate opening of the LDW deluge lines (feeding off the GDCS pools), thus preventing premature flooding.

The BiMAC design makes it functional immediately upon opening the deluge lines. Thus, there is no need to preflood the LDW, and the detailed design of the deluge lines valve activation system is based upon detecting melt arrival onto the LDW floor. This activation system is accessible both automatically as well as by operator action, and the required reliability is set at less than 1×10^{-3} failure per demand.

There is no ESBWR requirement to initiate drywell sprays and the emergency procedure guidelines (EPGs) do not use drywell sprays. They only appear as options in the severe accident management guidelines (SAMGs). Section 19.2.3.3.8 further discusses spray usage.

Section 21.4 of the ESBWR PRA report describes the detailed probabilistic framework, quantification of EVE loads, quantification of fragility to EVE, and prediction of failure probability caused by EVE. The results of the studies on pedestal loads and fragility for 1- and 2-meter-deep highly subcooled pools, taken to bound loads from shallow, saturated pools, indicate a large margin to failure, thus suggesting that in 99 percent of the Class I severe accidents in the ESBWR, pedestal failure by an EVE is PU.

The following are the principal components of such a conclusion:

- an accident management strategy and related hardware features that prohibit large amounts of cold water from entering the LDW before RPV breach

- the physical fact that premixtures in saturated water pools become highly voided and thus unable to support the escalation of natural triggers to thermal detonations

- reactor pedestal and BiMAC structural designs that are capable of resisting explosion load impulses of magnitudes in the hundreds of kPa-s

The remaining 1 percent refers to Class I accidents with deep (i.e., greater than 1.5 meters), subcooled water pools that constitute about 1 percent of the CDF. For such pools, although not considered in any detail, an appropriately conservative position would be that "integrity of both the liner and the concrete structure could be possibly compromised." Similar conclusions are drawn for the BiMAC function. The 1.5-meter demarcation for the "deep" water pool was selected because of the position of the hatch door, combined with a collective judgment aimed to leave out ranges of conditions that GEH did not believe could be reasonably captured by current capabilities and experience.

19.2.3.3.5.4 Conclusion

The staff concludes that in-vessel steam explosions are not a threat to the ESBWR containment based on the findings of the SERG. The staff finds the assumption that the occurrence of the flooded LDW at RPV failure leads directly to CF to be acceptable and conservative.

GEH states that the frequency of a flooded LDW at the time of reactor vessel failure is on the order of 10^{-10}/yr. This provides a sufficient basis to conclude that the frequency of an EVE leading to CF has been reduced to an acceptably low value and is therefore acceptable.

GEH performed analyses to determine the capability of the ESBWR containment to withstand EVEs for essentially all other cases (with LDW water levels below 1.5 meters and saturated water), even though failure in these cases is deemed PU. The staff previously performed separate analyses for the ABWR design to justify a similar conclusion for that design. (See ERI/NRC 93-203, "An Assessment of Ex-Vessel Fuel-Coolant-Interaction Energetics for the General Electric Advanced Boiling Water Reactor," letter dated March 12, 1993, Richard Borchardt, NRC, to Patrick Marriott, GEH.)

19.2.3.3.6 Containment Bypass

In SECY-90-016, the staff concluded that a special effort should be made to eliminate or further reduce the likelihood of a sequence that could bypass the containment. In SECY-93-087, the staff stated that vendors should make reasonable efforts to minimize the possibility of bypass leakage and their containment designs should account for a certain amount of bypass leakage.

19.2.3.3.6.1 Suppression Pool Bypass

The ESBWR PRA evaluates suppression pool bypass pathways. These potential pathways for the release of radioactive material do not receive the benefits of suppression pool scrubbing.

19.2.3.3.6.1.1 Logical Process Used to Select Important Design Features

GEH systematically reviewed the core cooling features that could prevent or mitigate containment bypass to determine their contribution to total CDF. The applicant identified those features that would increase the calculated CDF by more than a factor of 2, whether they failed or were not included in the design as important features. These features are evaluated below.

Drywell-Wetwell Vacuum Breakers

The PRA evaluates the consequence of a vacuum breaker failing to close or inadvertently remaining open.

Redundant MSIVs

If both MSIVs in any one main steamline fail to close, there will be a large bypass pathway, as compared to other potential bypass pathways, from the RPV to the turbine building. Therefore, the failure of two MSIVs to close in any one steamline would result in a higher consequence from a given postulated event. Depending on the event, the dual failure could result in a substantial offsite dose consequence.

Design and Fabrication of the SRV Discharge Lines

The discharges of the SRVs are piped downward through the drywell/wetwell vent wall and only emerge into the suppression pool below the pool surface. This configuration minimizes the potential for bypass of the suppression pool as a result of a break in one of these lines.

Normally Closed Sample Lines and Drywell Purge Lines

The sample lines and drywell purge lines are normally closed during plant power operation. If one or more of these lines are open when an event initiates, a potential bypass path can exist. Depending on the event and the size and number of lines open, a substantial fission product release could result in a significant increase in the consequences of a given event.

Diverse Reactor Water Cleanup System Isolation Valves

The probability of not isolating an RWCU line break outside containment is very low because of the inclusion of three automatic diverse isolation valves (in addition to a remote manual shutoff valve). Even though the exposed structures and safety-related equipment are designed for the loads and environment that could result from an unisolated break, there is some potential for failure. Furthermore, there is some potential that the operator will not properly control the reactor vessel water level during the recovery phase.

Other Less Important Plant Features

The applicant judged several plant features treated in the analysis to be much less important than those discussed above. As noted in the PRA, these include piping dimensions, the level of water in the suppression pool, the closing of the turbine bypass valve, the instrument check valves, and reliable seating of redundant feedwater and SLC check valves.

19.2.3.3.6.1.3 Conclusion

Release categories BOC, BYP, and OPVB bypass the suppression pool. Their combined frequency contributes 0.6 percent of the CFs and their risk contribution is 0.6 percent of the 72-hour societal risk within 10 miles. This is significantly less than 10 percent of the total offsite risk from internal event sequences and therefore does not present an undue offsite risk. Nevertheless, the staff requested further information on vacuum breaker performance in RAIs 19.2-6, 19.2-10, and 19.2-11. **RAIs 19.2-6, 19.2-10, and 19.2-11 are being tracked as open items.**

In SECY-90-016, the staff stated that containment venting should be delayed for approximately 24 hours following the onset of core damage. The ESBWR design does not credit the use of containment venting for preventing CF. The analysis includes containment venting simply to mitigate the magnitude of radionuclide releases resulting from loss of CHR by forcing the pathway through the suppression pool (see also Section 19.2.3.3.6.2 of this report). In virtually all circumstances, containment venting would not be initiated within the first 24 hours of core damage, as the containment pressure load at 24 hours would be still under the expected containment ultimate pressure capability.

The staff concludes that GEH performed a relatively complete analysis to facilitate an understanding of the capability of the ESBWR containment to accommodate a range of bypass conditions.

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.2.3.3.6.2 Containment Vent Design

The system called upon in the ESBWR EPG to control containment pressure is the CIS. This particular operational usage is referred to as the manual containment overpressure protection subsystem.

The ESBWR CIS design includes ventlines from the suppression chamber air space connected to the rooms directly below the suppression pool. In the event that CHR fails or CCI continues unabated, these CIS lines are opened under manual control to vent the wetwell gas space to the environment. This forces the higher pressure drywell gases to transfer to the lower pressure wetwell through the open wetwell-drywell vent paths, all of which go through the suppression pool water.

In a core damage event initiated by a transient in which the vessel does not fail, fission products are directed to the suppression pool via the SRVs, ICS, or PCCS, scrubbing any potential release. After RPV failure, the fission products are carried into the pool directly when the pressure differential is sufficiently large to activate the wetwell-drywell vents.

The vent is included in the PRA MAAP model by reflecting expected operator guidance to open a 2-inch line followed by a 12-inch line as needed to control pressure rise. The vent is not credited in the base sequences, but its effect is evaluated separately in Section 8.3 the PRA report. For modeling purposes, it is assumed that venting would occur only if containment pressure reached 90 percent of the ultimate pressure capability. Depending on the sequence details, this limit would be reached after 24 hours into the accident.

GEH believes that this arrangement for venting is satisfactory. The line sizes are adequate and the system has the requisite monitoring and control capabilities.

19.2.3.3.7 Equipment Survivability

SECY-93-016 and SECY-93-087 require that a survivability evaluation consider “credible” severe accidents. Similarly, 10 CFR 50.34, “Contents of Construction Permit and Operating License Applications; Technical Information,” requires that equipment survivability consider an accident with the release of hydrogen generated by the equivalent of a 100-percent fuel-clad metal-water reaction.

Section 8D of the PRA report presents the equipment survivability analysis for the ESBWR. Equipment survivability is evaluated to demonstrate that necessary components and instrumentation will be functional in the severe accident environment so that the plant may be placed in a controlled, stable state.

19.2.3.3.7.1 Equipment and Instrumentation Necessary to Survive

The ESBWR severe accident functional requirements are based on the conservative assumption that all severe accident scenarios result in RPV failure and that recovery of failed equipment is not credited. That is, if equipment is failed or unavailable at any time during the accident sequence, it will not be repaired or made available. Only those components within the containment boundary are subject to the severe accident environment. It is from this perspective that the mitigating functions necessary to place the ESBWR in a stable, controlled configuration have been considered. These functions include cooling of corium debris bed (LDW), cooling of corium debris bed (UDW), containment isolation, containment pressure control by heat removal or venting, combustible gas control, and postaccident monitoring.

Table 19.2-1 summarizes the plant systems that are required to carry out severe accident functions. The table also lists the systems’ components that are subject to the severe accident environment.

Table 19.2-1 System Functions and Monitored Variables Needed After a Severe Accident (from Table 8D2-1 of the PRA)

Function	Monitored Variables
Cooling of Debris Bed (LDW)	LDW Temperature Deluge Valve Status Indication Drywell Air Temperature GDCS Tank Water Level Drywell Sump Level
Cooling of Debris Bed (UDW)	Drywell Air Temperature
Containment Isolation	Drywell Pressure Isolation Valve Position
Containment Pressure Control: Heat Removal	Drywell Pressure Wetwell Pressure Drywell Air Temperature
Containment Pressure Control: Venting	Drywell Pressure Wetwell Pressure
Combustible Gas Control	Drywell/Wetwell H_2 Concentration Drywell/Wetwell O_2 Concentration
Containment Water Level	Suppression Pool Level Drywell Sump Level
Containment Radiation Intensity	Containment Area Radiation Monitoring
Noble Gas and Effluents at Potential Release Points	Environment Release Point Monitoring

19.2.3.3.7.2 Severe Accident Environmental Conditions

The applicant performed MAAP simulations to predict containment conditions for three representative accident sequences (i.e., transient with and without reactor depressurization and no coolant injection and a medium LOCA in liquid line with no coolant injection, representing a low and a high reactor pressure and a LOCA sequence, respectively); conditions for a fourth sequence (main steamline break with no core injection, representing a 100-percent fuel clad-coolant interaction sequence) were calculated using conservative simplifying assumptions. Then, GEH developed composite curves of containment pressure and temperature over a 24-hour period to represent bounding severe accident conditions. The applicant estimated radiation levels after a severe accident using a simplified one-compartment model. It was assumed that releases of 100 percent of the core noble gases and 50 percent of the core halogens were instantaneous at the start of the accident. All noble gases and halogens were assumed airborne for the full calculation time period with no credit taken for suppression pool scrubbing or other removal processes, either natural or otherwise (leakage or purging).

The analyses showed that the bounding pressure curve levels off at approximately 0.62 MPa at 24 hours after onset of core damage. The calculated bounding UDW region temperature history indicates that, except for a short period, it does not exceed 660 K and subsequently remains below 560 K for the duration of the scenario. Based on these results, GEH indicated that reasonable assurance is provided that the integrity of the UDW electrical penetrations will be maintained at bounding conditions of 644 K and 1.025 MPa

GEH will provide the bounding radiation environment in the next DCD revision to this document.

It is anticipated that the integrated radiation dose within 24 hours that is associated with severe accident conditions will not exceed equipment design bases.

19.2.3.3.7.3 Conclusion

The applicant carried out a systematic evaluation to evaluate the capability of the equipment necessary to survive in a severe accident environment in the ESBWR and to demonstrate reasonable assurance of operability. In doing so, GEH considered physical location, design or qualification in comparison to the severe accident environment, timing of the required equipment function, nature of the required equipment function, duration of the severe accident condition, and material properties. The severe accident environment was established by evaluating credible representative severe accident scenarios from the PRA, as well as a nonmechanistic 100-percent fuel-clad metal-water reaction. The evaluation was for a 24-hour period after onset of core damage.

Table 8D.4-2 of the PRA summarizes the evaluation of severe accident equipment capability. The evaluation provided reasonable assurance that the ESBWR equipment necessary to achieve a controlled, stable plant condition will function over the time span in which it is needed.

19.2.3.3.8 Non-Safety-Related Containment Spray

As discussed above in Section 19.1.3.2.1, “Drywell Spray Function of the FAPCS,” the SAMGs will not include the use of the drywell spray system, and the PRA Level 2 and Level 3 analysis does not include drywell sprays.

No detailed designs for the spray systems have been put forward. Several statements in the DCD imply that there will be interlocks that must be overridden before the sprays can be utilized.

The PRA does not credit any benefit of the containment spray system on fission product releases.

19.2.4 Containment Performance Capability

The purpose of this section is to provide the staff’s assessment of the ESBWR containment structural performance to resist loads induced by postulated beyond-design-basis severe accidents. The staff reviewed applicable sections of DCD Tier 2 (Revisions 0, 1, 2 and 3), Section 3.8.1, “Concrete Containment”; Section 3.8.2, “Steel Components of the Reinforced Containment”; and Section 6.2.5.4, “Containment Over-pressure Protection” which provide the applicant’s assessment of the containment structural capacity to withstand overpressurization resulting from a 100-percent fuel clad-coolant reaction, in accordance with Service Level C of the ASME Boiler and Pressure Vessel Code (ASME Code) or factored load limits. GEH also performed a PRA of the ESBWR containment structural capability against loads induced by the more likely severe accident scenarios. GEH described the probabilistic assessment of the containment capacity in Section 19 of DCD Tier 2. The staff reviewed the applicable sections of DCD Tier 2, Revisions 0 and 1, related to containment SMA, containment performance against overpressurization, and containment phenomenological event analysis for EVE and DCH. The staff also reviewed applicable sections of the PRA report, Revisions 0 and 1.

Based on the review and evaluation of the DCD and the supporting PRA report, the staff determined the adequacy of the applicant’s assessment of the containment structural capacity to withstand loads induced by the more likely severe accident scenarios, which were provided to GEH. GEH subsequently issued DCD Revision 4 and PRA Revision 2, which included a revision to the severe accident containment structural capacity assessments contained in previous versions of the DCD and PRA report. This section of the report describes the GEH containment structural performance assessment and the associated staff evaluation.

19.2.4.1 Regulatory Criteria

The following are the relevant regulations and regulatory guidance that the staff used for performing this review:

- General Design Criterion (GDC) 16, “Containment Design,” relates to the capability of the containment to act as a leaktight membrane to prevent the uncontrolled release of radioactive effluents to the environment.

- GDC 50, “Containment Design Basis,” relates to the containment being designed with sufficient margin of safety to accommodate appropriate design loads.

- Appendix B, “Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities” relates to the quality assurance criteria for nuclear power plants.

- 10 CFR 52.47(a)(1)(vi) requires that a DC application contain the proposed ITAAC that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, a plant that incorporates the DC is built and will operate in accordance with the DC, the provisions of the Atomic Energy Act, and the NRC’s regulations.

- 10 CFR 52.79(c) requires that a COL application contain the proposed inspections, tests, and analyses, including those applicable to emergency planning, that the licensee shall perform, and the acceptance criteria that are necessary and sufficient to provide reasonable assurance that, if the inspections, tests, and analyses are performed and the acceptance criteria met, the facility has been constructed and will operate in conformity with the COL, the provisions of the Atomic Energy Act, and the NRC’s regulations.

- 10 CFR 50.44 requires the containment integrity to withstand pressurization induced by an accident that releases hydrogen generated from fuel clad-coolant reaction accompanied by hydrogen burning. In particular, 10 CFR 50.44(c)(5) requires that an analysis using an analytical technique acceptable to the staff be performed to demonstrate the containment integrity to withstand internal pressurization from an accident that releases hydrogen generated from the 100-percent fuel clad-coolant reaction

- RG 1.70, Revision 3, “Standard Format and Content of Safety Analysis Report for Nuclear Power Plants,” issued November 1978, provides guidance for meeting the 10 CFR 50.44(c)(5) requirement and specifies the following:

  - steel containments meet the requirements of the ASME Code (edition and addenda as incorporated by reference in 10 CFR 50.55a(b)(1)), Section III, Division 1, Subsubarticle NE-3220, Service Level C Limits, considering pressure and dead load alone (evaluation of instability is not required)

– concrete containments meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 2, Subarticle CC-3720, Factored Load Category, considering pressure and dead load alone

At a minimum, the specific ASME Code requirements set forth for each type of containment will be met for a combination of dead load and an internal pressure of 45 psig.

- 10 CFR 52.47(a)(1)(v) requires that the applicant provide a description of the design-specific PRA and its results.

- SECY-93-087 and the Commission's SRM provide guidance for meeting the deterministic CPG in the evaluation of the passive ALWRs as a complement to the CCFP approach. The SECY-93-087 expectation with respect to the deterministic containment performance assessment is described as follows:

  The containment should maintain its role as a reliable, leaktight barrier (e.g., by ensuring that containment stresses do not exceed ASME Service Level C limits for metal containment or factored load category for concrete containments) for approximately 24 hours following the onset of core damage under the most likely severe accident challenges, and following this period, the containment should continue to provide a barrier against the uncontrolled release of fission products.

- SECY-93-087, Section II.N, and the Commission's SRM also provide guidance for a sequence-level SMA. PRA insights will be used to support a margin-type assessment of seismic events. A PRA-based SMA will consider sequence-level HCLPFs and fragilities for all sequences leading to core damage or CFs up to approximately 1.67 the ground motion acceleration of the design-basis SSE.

19.2.4.2 <u>Summary of Technical Information</u>

In DCD Tier 2, Section 3.8, GEH described the physical characteristics of the concrete containment for the ESBWR plant. The containment is a reinforced concrete structure with steel liner, and the containment pressure boundary consists of a foundation mat, cylindrical walls, RPV pedestal, suppression pool slab, girder-spanned top slab, and steel drywell head. Other internal structures that may be subject to severe accident loads include those located in the LDW and UDW areas, the vent wall separating the suppression pool, and the diaphragm floor supporting the GDCS pools. The pressure capability of the drywell head, as well as the major containment penetrations (equipment hatch, personnel airlock, wetwell hatch), including penetrations for process piping and electrical cables, may also be affected by the severe accident loads.

The containment structure is designed to resist various combinations of dead loads; live loads; environmental loads, including earthquakes and those resulting indirectly from wind and tornadoes; normal operating loads; and loads generated by a postulated LOCA. The primary function of the containment structure is to provide the principal barrier to control potential fission product releases to the environment under a postulated LOCA. The ESBWR primary containment is designed to withstand a maximum pressure of 0.310 MPa (45 psig) and a design temperature of 340 ^{o}F.

19.2.4.2.1 10 CFR 50.44 Requirement

In DCD Tier 2, Section 6.2.5.4, GEH provided a deterministic assessment of containment structural capability against overpressurization. GEH performed its analysis to address 10 CFR 50.44 utilizing the guidance of RG 1.70, Revision 3. GEH stated that the pressure capability of the containment's limiting component is higher than the pressure that results from assuming a 100-percent fuel clad-coolant reaction. GEH estimated the Level C pressure capability of the steel components of major penetrations (drywell head, equipment hatch, personnel airlock, and wetwell hatch); Table 6.2-46 presents the resulting Level C pressure capacities for these components. The controlling component, which has the lowest calculated Level C pressure capacity, is the drywell head, with a calculated Level C pressure capacity equal to 1.182 MPa. Circumferential buckling in the knuckle region of the torispherical head is identified to be the failure mode.

In Appendices 19B and 19C to DCD Tier 2, Revision 1, GEH summarized the fragility analysis to determine the structural capability of the ESBWR containment to withstand overpressurization.

In Section 8 of the PRA report, Revision 1, GEH described the analysis and evaluation of the ability of the ESBWR containment to address the system-related containment challenges associated with potential combustible gas deflagration, overpressurization, and bypass. The ESBWR design employs an inerted containment. The GEH radiolytic oxygen concentration analysis, which assumes a 100-percent fuel clad-coolant reaction, showed that the time for the oxygen generation to increase to the deinerting value of 5-percent containment volume, following a severe accident, is significantly greater than 24 hours. Therefore, there is sufficient time for implementation of severe accident management actions. The ESBWR primary containment design accident pressure is 0.310 MPa (45 psig), the minimum requirement of RG 1.7. GEH concluded that the CF caused by combustible gas deflagration is unrealistic.

In Appendix 19B to DCD Tier 2, Revision 4, GEH included an assessment using a deterministic analysis to demonstrate Level C pressure capability of the RCCV and major penetrations. This analysis was based on detailed three-dimensional finite element modeling using the ABAQUS/ANACAP-U concrete model and specified design values for material properties. GEH assessed the Level C or factored load pressure capability of the containment structure to be no less than 0.987 MPa gauge (143 psig) generated from a 100-percent fuel clad-coolant reaction, taking into account temperature effect on the material strength.

19.2.4.2.2 SECY-93-087 Deterministic Containment Performance Expectation

GEH addressed the SECY-93-087, Section I.J, expectation regarding the deterministic containment performance assessment in DCD Tier 2, Revision 1, Section 19.3.2.4. GEH performed its assessment in accordance with the guidance of RG 1.70, Revision 3. GEH determined the Level C pressure capability of the containment to be 1.182 MPa, which is governed by the buckling failure of the drywell head. However, GEH did not provide the pressure and temperature time histories for the more likely accident scenarios, as expected by SECY-93-087, and did not address the effect of elevated temperature on the material properties used in the Level C calculation.

GEH included the revised SECY-93-087 analysis in its analysis for the deterministic containment structural performance assessment as provided in Appendix 19B to DCD Tier 2, Revision 4. The containment pressure and temperature distributions resulting from the more likely accident scenarios (the top sequences accounting for 97 percent of CDF were considered), were determined to be enveloped by the sequence T_nDP_nIN_TSL, a Class III sequence. GEH assessed the Level C or factored load pressure capability of the containment structure to be no less than 0.62 MPa gauge (90 psig) resulting from more likely severe accident challenges, taking into account the temperature effect on the material strength.

19.2.4.2.3 Probabilistic Containment Performance Assessment

GEH performed the containment performance assessment for overpressurization and developed the containment pressure fragility. Appendices 19B and 19C to DCD Tier 2, Revision 1, describe the GEH assessment. Section 8 of Revision 1 of the PRA report provides a detailed assessment of the system-related containment challenges. The detailed fragility analysis, described in Appendix B.8 to Revision 1 of the PRA report, evaluates the ultimate containment pressure capacity of the various structural components comprising the containment pressure boundary and identifies the controlling component and failure mode. The pressure capacity of the controlling component was used in developing the containment fragility, which was assumed to have a lognormal distribution. The applicant performed the evaluation of containment ultimate pressure strength using a two-dimensional axisymmetric ANSYS analysis of the RCCV at ambient temperature. The ANSYS analysis results were then reduced by 10 percent to represent the ultimate pressure strength at a typical accident temperature, which GEH defines as 533 K (500 °F). The applicant performed a separate analysis for the drywell head, which is a carbon steel SA-516, Gr. 70 torospherical shell structure, based on several empirical equations. GEH concluded that the ultimate pressure capability of the containment is limited by the drywell head. The pressure capability of the drywell head is estimated to be 1.204 MPa (174 psi) at 500 °F.

GEH estimated the median pressure capacity of the drywell head in the PRA report, Revision 1, Section B.8.3, to be 1.623 MPa at 500 °F using an empirical equation. The applicant further determined a composite logarithmic standard deviation of 0.16 about the median pressure. Based on the lognormal distribution, GEH stated that the containment pressure strength at 2 logarithmic standard deviations below the mean is 1.111 MPa, or 3.58 times the design pressure of 0.31 MPa (45 psi).

In Appendix 19C to DCD Tier 2, Revision 4, the containment fragility for internal pressurization was re-performed using more detailed ABAQUS/ANACAP-U three-dimensional finite element model. The analysis made use of the material properties of the in situ structure at the time of the accident, failure criteria, or limit states in establishing the containment pressure capacity. Median capacity was calculated by setting all parameters to their median values.

The applicant assessed the randomness in material properties and failure criteria by first identifying those parameters that are likely to have a significant effect on the analysis results and evaluating the effect of variations in these parameters using a 95-percent confidence value of the specific parameter, assuming a normal distribution while keeping all other parameters at the median values. A lognormal distribution characterized the failure pressure. Therefore, the applicant estimated the uncertainty in the failure pressure caused by the randomness of a parameter using the relation, beta = $\mathrm{Ln}(P_{95}/P_m)/(-1.645)$. GEH then aggregated the uncertainty for all parameters using the square root of the sum of the squares (SRSS) method.

GEH assessed modeling uncertainty (e.g., mesh fidelity, element formulations robustness of the constitutive models) based on past experience and analyst judgment. The uncertainty was further increased to account for the various thermal conditions. The modeling uncertainty was then combined with the random uncertainty using SRSS, resulting in the containment pressure fragility.

GEH addressed the containment bypass using a screening evaluation, as documented in Section C.8 of the PRA report, Revision 1. The GEH screening analysis indicated that there were no penetrations that require isolation to prevent significant offsite consequences. GEH determined that the probability of the bypass failure mode is dominated by a common isolation signal failure probability, resulting in a calculated frequency of containment bypass about 4 orders of magnitude lower than the TSL release.

GEH also addressed the containment phenomenological challenges. DCD Tier 2, Revision 1, Section 19.3 summarizes the GEH assessment of the ESBWR containment capability against phenomenological challenges induced by DCH, EVE, and BMP. Chapter 21 of the PRA report, Revision 1, details the containment phenomenological analysis.

DCH occurs when high-velocity steam from an RPV high-pressure blowdown impinges upon melt debris already released onto the LDW floor, thus creating a finely atomized melt mixture. The atomized hot melt is then dispersed into and heats up the UDW. In Section 21.3 of the PRA report, Revision 1, GEH stated that the set of accidents that could lead to DCH involve core degradation and vessel failure at high primary system pressure and the probability for such events to occur is very small (i.e., $2.8x10^{-9}$). The Level 1 PRA also indicates that high-pressure accidents contribute only about 1 percent of the CDF. Therefore, GEH concluded that a DCH event in the ESBWR is PU, and the DCH events discussed in the same section were categorized as remote and speculative. GEH also stated that the potential for LDW liner failure can be induced by a DCH event. In addition, according to the summary of results of severe accident sequence analysis presented in Table 8.3-3 of the PRA report, Revision 1, the concrete ablation 24 hours after onset of core damage is 0.1 meter or less.

EVE events are energetic FCIs which are triggered by melt released from lower RPV head breach falling into a preexisting subcooled water pool in the LDW cavity. EVE events develop pressure impulses (the time-integral of the pressure load), which could damage LDW structures, such as the pedestal, and the BiMAC device.

In Section 21.4 of the PRA report, Revision 1, GEH described the containment and BiMAC performance against EVE. The relevant structures that may be subjected to potential damage are the 2.5-meter-thick reinforced concrete reactor pedestal and the BiMAC device. The conditions for EVE are the presence of water and lower RPV pressure (low pressure, defined as the RPV pressure less than 1 MPa). In the GEH analysis, the water depth is divided into three categories—high (H greater than 1.5 meters), medium (H between 0.7 and 1.5 meters) and low ((H less than 0.7 meter), where H is the depth of the subcooled water pool in the LDW cavity. For the high-level depth of the subcooled water pool, which involves only 0.9 percent of CDF, the failures of the structures involved are considered possible. For the other two water depths, which constitute 99 percent of CDF, GEH performed DYNA-3D analyses of the pedestal and BiMAC, concluding that the pedestal is capable of resisting pressure impulses of over 500 kPa-s, and the BiMAC can sustain a pressure impulse of over 100 kPa-s, the maximum pressure impulses induced by the EVE events. Therefore, 99 percent of the low-pressure sequences (Class I) can be excluded for the EVE evaluation. Based on the analysis results, GEH concluded that, for all but 1 percent of the CDF, violations of the containment integrity and BiMAC function are considered PU.

The BMP events involve any amount of melt debris that is not coolable, and the decay power is split between the upwards (into water) and downwards (into concrete) directions. Both high-pressure and low-pressure scenarios need to consider BMP. In Section 21.5 of the PRA report, Revision 1, GEH described the design of the BiMAC device, especially the section of a refractory ceramic material that serves as a protective layer, eliminating ablation by superheated meats and preventing basemat penetration of the molten core debris for a minimum of 24 hours, and hence, the CF.

19.2.4.2.4 Drywell Head

In DCD Tier 2, Revision 0, Section 3.8.2, GEH described the drywell head as a removable steel torispherical shell structure that covers the opening in the containment's UDW top slab, directly above the RPV. The head is designed for removal during reactor refueling, using the reactor building crane.

DCD Tier 2, Revision 0, Section 6.2.5.4.2, presents a detailed deterministic analysis of the Level C internal pressure capacity for the drywell head at ambient temperature. This estimate is based on a design equation proposed by (Equation (6.2-2) in Galletly, "A Simple Design Equation for Preventing Buckling in Fabricated Torispherical Shells under Internal Pressure". The Galletly equation was qualified based on a comparison to 43 test results. GEH had previously performed a statistical analysis of the test data on which Equation (6.2-2) is based and documented it in the ABWR DCD. GEH identified the critical location to be the knuckle region of the torispherical geometry. The calculated Level C pressure capacity is equal to 1.182 MPa; circumferential buckling of the knuckle region is identified to be the failure mode.

GEH reevaluated the Level C capacity of the drywell head in Appendix B to Chapter 19 of DCD Tier 2, Revision 1 by calculating the Level C/factored load capacity in accordance with ASME Code, Section III, Divisions 1 and 2. The buckling failure of the head shell was precluded because of a low diameter/thickness ratio (D/t = 260), which was confirmed by a detailed finite element analysis. The applicant determined the governing pressure for the drywell head to be 1.033MPa gauge (150 psig), which is controlled by the lower flange plate of the anchorage.

In Appendix B.8 to the PRA report, Revision 1, GEH presented a fragility analysis to determine the structural capability of the drywell head under internal pressure and temperature loading. GEH analyzed the pressure capacity of the head shell under ambient temperature, based on the Equation (B.8-1) in Shield and Drucker, "Design of Thin-Walled Torispherical and Toriconical Pressure-Vessel Heads" for plastic yielding failure mode and the Galletly Equation (B.8-3). GEH determined that the pressure capacity of the head shell is governed by the Shield and Drucker equation (B.8-1) for plastic yielding failure mode. GEH stated that, during various accident conditions, the ESBWR containment could be challenged by high temperature, with a typical accident temperature about 533 K (500 °F). To obtain a more realistic estimate of the structural strength of the head shell, GEH increased the minimum yield strength of the shell material SA-516, Gr. 70 at 533 K by 10 percent. On the basis of the Shield and Drucker Equation (B.8-1), GEH estimated the ultimate pressure capacity of the drywell head at 500 °F to be 1.204 MPa (174 psig) with plastic yielding as the failure mode. GEH also stated that the containment ultimate pressure capability is limited by failure of the drywell head.

GEH further stated that a separate equation (B.8-10) by Galletly (Galletly and Radhamohan, "Elastic-Plastic Buckling of Internally-Pressurized Thin Torispherical Shells" and Galletly and Blachnut, "Torispherical Shells Under Internal Pressure-Failure Due to Asymmetric Plastic

Buckling or Axi-symmetric Yielding" provided a lower estimate of the shell pressure capacity than did the Shield and Drucker Equation (B.8-1). Therefore, the applicant used the Galletly Equation (B.8-10) to estimate the median pressure capacity of the drywell head, which is 1.623 MPa at 500 °F. GEH also estimated a composite logarithmic standard deviation of 0.16 for the shell material SA-516, Gr. 70. Based on the lognormal distribution, GEH stated that the containment pressure strength at 2 logarithmic standard deviations below the mean is 1.111 MPa, or 3.58 times the design pressure of 0.31 MPa (45 psig), governed by the plastic yielding of the drywell head shell.

The applicant later revised the fragility analysis in Appendix C to Chapter 19 of DCD Tier 2, Revision 4 based on a detailed finite element model. This analysis determined that the bending or prying deformation response in the bolted flanges stretches bolts to yield, leading to the failure of the head, according to the established failure criteria. The applicant determined the 95-percent confidence value for the failure pressure to be 1.443 MPa gauge ($4.65P_d$) at 260 °C (500 °F).

19.2.4.2.4.1 External Water Presence and Temperature during Severe Accidents

The drywell head seals the cylindrical top portion of the UDW. The outside surface of the drywell head is immersed in a water pool during normal operation. The function of the water pool is to provide shielding for radiation. The water pool is isolated from other cooling pools (e.g., PCCS/IC pools). The pool is periodically replenished during normal operation. The presence of this water pool limits the temperature increase through the thickness of the drywell head, condenses steam accumulated on the inside surface of the head, and provides significant scrubbing of the fission products released through failed drywell head seals. GEH stated in Section 21.3.4.4 of the PRA report, Revision 1, that bounding estimates of this process yield internal drywell temperatures of less than 450 K. GEH also expected that this cooling by the water pool would be effective in the long term and sufficient to accommodate the thermal loads from the hot UDW atmosphere, as it may develop during a DCH event.

19.2.4.2.5 Reinforced Concrete Containment Vessel

In DCD Tier 2, Revision 0, Section 3.8.1, GEH described the RCCV as a cylindrical reinforced concrete structure with an internal welded steel plate liner. The liner is made of carbon steel, except for the wetted surfaces of the suppression chamber and GDCS pools, where stainless steel or carbon steel with stainless steel cladding will be used. The RCCV is surrounded by and structurally integral with the reinforced concrete reactor building through the floor slabs, the IC/PCC pools, and the service pools used for storage of the dryer/moisture separator and other components.

In DCD Tier 2, Revision 0, Section 6.2.5.4.2, GEH evaluated the Level C (factored load) pressure capability of the RCCV using the liner strain limits for factored load category specified in ASME Code, Section III, Division 2, Table CC-3720. GEH estimated the maximum liner strain from a nonlinear finite element analysis of the containment concrete structure, including liner plates, for internal pressure loading. No reference is provided for the analysis. GEH stated that the maximum strain is only 0.165 percent in tension when the internal pressure reaches 1.468 MPa, which is higher than the 1.182 MPa pressure for the drywell head.

In Appendix B.8 to the PRA report, Revision 1, GEH presented an ANSYS axisymmetric finite element analysis of the RCCV subject to internal pressure and dead load at ambient temperature. The applicant scaled down the ultimate pressure capability values resulting from the ANSYS analysis by 10 percent to represent the pressure capability of the RCCV at 533 K (500 °F). Table B.8-2 summarizes the calculated pressure capacities of various RCCV components. The ANSYS analysis determined the pressure capacity of the RCCV to be 1.468 MPa at ambient temperature. The failure mode is identified as a shear failure of the suppression pool slab at the junction with the containment wall.

The applicant revised both the Level C/factored load and fragility analyses for the RCCV, as discussed above, in Appendices 19B and 19C to DCD Tier 2, Revision 4. The new analyses were based on a separate three-dimensional ABAQUS/ANACAP-U finite element model and considered the temperature effect on material properties. Level C pressure capacity was not provided. However, to address SECY-93-087, GEH performed an analysis to calculate the RCCV response to the internal pressure of 0.62 MPa gauge (90 psig) corresponding to the more likely severe accident conditions. The induced stresses and strains within the RCCV were found to be less than the Level C (factored load) allowable limits. If the internal pressure is increased to 0.987MPa gauge (143 psig), corresponding to the 100-percent fuel-coolant reaction pressure, the liner in the UDW wall connection with the top slab will undergo 0.72-percent tensile strain, exceeding the factored load allowable of 0.3 percent. DCD Tier 2, Revision 4, Section 19B.2.3, states, "...if the thermal stress is included, then the liner strain is within the factored load limit…."

19.2.4.2.5.1 Severe Accident Temperature Loads

Section 8.3 of the PRA report, Revision 1, describes the temperature loads for the RCCV induced by the more likely severe accidents. This section provides the temperature transient time histories for the RCCV for two system-initiated sequences—T_nDP_nIN_TSL and T_nIN_nCHR_FR. T nIN_TSL represents a sequence in which no short- or long-term injection is available, with TSL being the only mode of fission product release. T_nIN_nCHR_FR denotes the sequence in which both vessel injection and CHR functions are unavailable. Containment venting needs to be implemented to limit the containment pressure rise and to control the radionuclide release point. For both sequences, Section 8.3 of the PRA report, Revision 1, only provides the temperature time histories for the LDW, which show the steady-state temperature to be nearly 450 K (350 °F).

Another source for high temperature loading on the RCCV is from a DCH event. DCH is a phenomenological event postulated for high-pressure core melt ejection from an RPV lower head penetration failure. In DCD Tier 2, Revision 0, Section 19.3.3, and Section 21.3 of the PRA report, Revision 1, GEH characterized the potential for a DCH event to occur as remote and speculative. DCH events are not grouped in the category of the more likely severe accident scenarios for the ESBWR. In DCD Tier 2, Revision 0, Section 19.3.3, and Section 21.3 of the PRA report, Revision 1, GEH discussed in a hypothetical context a CF caused by DCH events. The applicant indicated that, in the event of an RPV failure at high pressure (above 1 MPa), the superheating of gases generated within a timeframe of 40 to 80 minutes following core uncovering can lead to temperature levels of 1000 K (1340 °F) in the upper RPV area. After taking credit for vent clearing from the UDW into the heat sink of the wetwell, the drywell temperature would be reduced to 800 K (980 °F). However, GEH pointed out that the necessary condition for a DCH event to occur requires that a minimum of two out of four ICS fail because of either water depletion on the secondary side or failure to open the condensate return valves. In addition, all 8 of the DPVs and 18 of the SRVs must fail. GEH indicated that it assessed the probability of such a combination of events to be 2.8×10^{-9}/yr. Therefore, GEH concluded that a DCH event is PU.

19.2.4.2.5.2 Environment Loads—Seismic: Estimates of Containment Seismic Fragility

In DCD Tier 2, Revision 0, Section 19.2.2.4, GEH summarized an SMA for Category I structures, including the RCCV. In DCD Tier 2, Revision 1, the bulk of the summary description in Revision 0 was removed; instead, GEH provided a brief description in DCD Tier 2, Revision 1, Section 19.2.3.5 (DCD Tier 2, Revisions 2 and 3, do not contain any update for Chapter 19), which also includes a table of the qualitative structural HCLPF capacities. Section 15 of the PRA report, Revision 1, provides the detailed description of both the method and resulting HCLPF values for Category I structures. The applicant determined the plant HCLPF value from the SSC HCLPF values using the MAX-MIN method.

In Section 15 of the PRA report, Revision 1, GEH described the SMA performed for Category I structures and presented respective HCLPF values, including the RCCV. The Zion method in NUREG/CR-2300, "PRA Procedures Guide," was applied to the seismic fragility calculations. The applicant calculated the seismic HCLPF for the containment to be 1.4g with the shear failure mode. The lowest HCLPF value for other structural components of RCCV is estimated to be 0.62g, controlled by channel deflection in the fuel assemblies. Thus, GEH determined the plant seismic HCLPF to be 0.62g.

The design SSE for the ESBWR is governed by the spectrum discussed in RG. 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," anchored at 0.3g (peak ground acceleration (PGA), and the North Anna Early Site Permit (ESP) site specific SSE spectrum. In accordance with the soil-structure interaction (SSI) analysis described in Appendix 3A to DCD Tier 2, generic sites with 0.3g RG 1.60 input typically result in higher structural responses than the North Anna ESP conditions for building structures, including the containment. Therefore, GEH used the RG 1.60 spectrum anchored at 0.3g PGA for the design seismic load calculation.

The applicant used the NUREG/CR-0098 median spectrum shape for fragility calculations and described various safety factors established between the NUREG/CR-0098 spectrum and the RG 1.60 spectrum. Table 15-3 of DCD Tier 2 presents the final fragility for the RCCV. GEH demonstrated that the ESBWR containment meets the SECY-93-087 expectation for the seismic margin assessment. The sequence-level HCLPF is at least 1.67 times SSE (0.5g PGA). The HCLPF value for the RCCV is 1.40g PGA, with the failure mode characterized as shear failure of the containment lower wall.

DCD Tier 2, Revision 4, and the PRA report, Revision 2, developed a performance-based design spectrum, which is the same as the single certified design spectrum at 9 hertz (Hz) and above. For lower frequencies, the applicant used a spectrum shape that bounds all the soil sites except Vogtle. Therefore, the performance-based design spectrum falls below the single certified design spectrum for frequencies below 9 Hz, which affects the HCLPF capacity calculations for soil sites.

Based on the performance-based design spectrum, GEH performed a PRA-based SMA, which determined that the sequence-level HCLPF is at least 1.67 times the SSE tied to the performance-based design spectrum.

19.2.4.2.5.3 Containment Liner - Failure of Pressure Containment Function during Severe Accident Loadings

In DCD Tier 2, Revision 0, Section 6.2.5.4.2, and Appendix B.8 to the PRA report, Revision 1, GEH discussed the structural capacity of containment liner plates when the internal pressure is as high as 1.468 MPa (215 psig). The maximum liner strains are found to be well within the ASME Code allowable for factored load category. GEH also stated that the liner plate analysis indicated no tearing at the severe accident pressure of 1.204 MPa (174 psig). The most significant effect of thermal loading on the liner is a potential buckling failure if the internal pressure-induced liner tensile stress is insufficient to overcome the thermal-induced compressive stress. Therefore, the potential for thermal-induced liner buckling can only be examined within the context of the containment pressure and temperature time histories associated with the more likely severe accident scenarios. GEH estimated that a typical severe accident temperature for the ESBWR component is 500 ^{o}F. At this temperature, GEH concluded that the ESBWR liner would not fail, given a containment internal pressure of 1.204 MPa.

In Appendix 19B to DCD Tier 2, Revision 4, the GEH new Level C analysis estimated that the liner strain will exceed the Level C allowable limit for internal pressure of 0.987 MPa gauge (143 psig) unless the thermal-induced compressive liner strain is included, which reduces the level of tensile strain in the liner.

For the fragility analysis, as documented in Appendix 19C to DCD Tier 2, Revision 4, the failure criteria for liner strain at a 95-percent confidence level was established at 2.04 percent at 260 °C (500 °F), and the corresponding 95-percent failure pressure for RCCV was calculated to be 1.317 MPa gauge (4.25P_d), governed by the liner tear at the RCCV wall connection with the top slab.

Although GEH characterized DCH events as unlikely accident scenarios, uncertainty about such event estimates is large. Therefore, GEH performed a reactor analysis to estimate the DCH-induced containment temperature and a structural analysis to evaluate the potential for thermal-induced liner failures. Based on the reactor analysis described in Section 21.3 of the PRA report, Revision 1, the DCH-induced UDW temperatures are estimated to be about 1000 K (1340 °F); however, for very short periods (less than 1 second), GEH estimated that the LDW could experience very high temperatures, up to 4000 K (6740 °F). A DYNA-3D analysis shows that a liner with concrete backing can sustain high temperatures up to 1650 K (2510 °F), and the calculated thermal strains are about 8 percent.

19.2.4.2.5.4 Penetrations - Failure of Pressure Containment Function during Severe Accident Loadings

In Sections 8.2 and B.8.2.2.2 of the PRA report, Revision 1, GEH discussed the major penetrations, such as the drywell head closure, equipment hatches, and personnel airlocks. The penetrations have a high potential for leakage under severe accident conditions. Leakage through fixed penetrations for process piping and electrical cables is assumed to be less likely. The seal performance depends mainly on temperature as well as the effect of thermal and radiation aging of seal materials. Test data for the sealing materials are used to qualify their performance under severe accident conditions. In addition, GEH presented a screening analysis to identify penetrations that could potentially lead to offsite consequences. Appendix C.8 to the PRA report, Revision 1, details the penetration screening analysis.

Appendices 19B and 19C to DCD Tier 2, Revision 4, provide Level C and fragility evaluations of equipment hatches and personnel airlocks, based on the new ABAQUS/ANACAP-U three-dimensional finite element models. The new analyses conclude that these main penetrations have much higher Level C and fragility in terms of the 95-percent values than do the RCCV and drywell head.

19.2.4.2.6 Reactor Cavity Structures

In DCD Tier 2, Revision 1, Section 19.3.4, and Sections 21.4 and 21.5 of the PRA report, Revision 1, GEH discussed the structural components that would be affected by potential EVEs and BMP. These include the reactor pedestal, reinforced concrete basemat, and BiMAC device. EVE is a postulated internal initiated event of energetic FCIs. An EVE is triggered as the core melt released from the failed RPV lower head falls into and traverses the depth of an already existing water pool on the LDW floor. The result of EVE events is energetic pressure pulses, with magnitudes in the kilo-bar range, that are potentially capable of loading major structures to failure when large quantities of melt react with highly subcooled water. The EVE loading is characterized by the impulse (the time-integral of the pressure) acting on the surface of a structure.

BMP events involve any amount of melt debris released onto the LDW floor that is not coolable. The decay power is split between the upward (into water) and downward (into concrete) directions. Both high-pressure and low-pressure scenarios need to consider BMP. The potential effect of BMP is CCI.

19.2.4.2.6.1 Reactor Cavity—Structural Performance under Ex-Vessel Steam Explosion Loadings

In DCD Tier 2, Revision 0, Section 19.3.4, and Section 21.4 of the PRA report, Revision 1, GEH discussed potential damage to structures caused by EVE loadings. The reactor cavity is enclosed by the reactor pedestal on the side and basemat on the bottom. Failure of the reactor pedestal, along with the steel liner on it, constitutes violation of the containment boundary. The GEH assessment includes using PM-ALPHA-3D to quantify the EVE loadings and an LS-DYNA3D analysis to determine the structural response of the pedestal and its liner. GEH concluded that steam-explosion-induced failures of the reactor pedestal and the steel liner are PU.

The conditions for EVE are the presence of water and lower RPV pressure (low pressure). The GEH analysis divides the water depth into three categories—high (H is greater than 1.5 meters), medium (H is between 0.7 and 1.5 meters) and low (H is less than 0.7 meter). H is the depth of the subcooled water pool in the LDW cavity, measured from the bottom of the reactor cavity. For the high-level depth of the subcooled water pool, which involves only 0.9 percent of CDF, failure of the affected structures is considered possible. For the other two water depths, which constitute 99 percent of CDF, GEH performed a DYNA-3D analysis for the pedestal and concluded that the pedestal is capable of resisting a pressure impulse of more than 500 kPa-s. For the high-water depth (H = 1.5 meter), there is a 2.2-meter gap between the top of water and the bottom of the pedestal penetration; therefore, it is unlikely that an EVE event could affect the penetration. On the basis of the analysis results, GEH concluded that, for all but less than 1 percent of CDF, violations of the containment integrity are considered PU.

19.2.4.2.6.2 BiMAC Device—Structural Performance under Ex-Vessel Steam Explosion Loadings

In DCD Tier 2, Revision 1, Section 19.3.4, and Sections 21.4 and 21.5 of the PRA report, Revision 1, GEH discussed the performance of the BiMAC in the LDW affected by EVE loading. The BiMAC device is comprised of thick-walled steel pipes protected by a sacrificial refractory layer of 0.2-meter-thick ceramic zirconia, with very high heat-resisting properties. GEH stated that the design of the BiMAC device, especially with the use of a refractory ceramic material as a protective layer, eliminates ablation by superheated melts. It prevents basemat penetration by the molten core debris for a minimum of 24 hours, maintaining containment integrity. In addition, the BiMAC cavity has a volume of about 400 percent of the full-core melt debris. Therefore, no possibility exists for the released melt to remain in contact with the reactor pedestal.

The GEH assessment included using PM-ALPHA-3D to quantify the EVE loadings and an LS-DYNA3D analysis to determine the structural response of the BiMAC device. GEH concluded that violation of the BiMAC function caused by EVE is PU.

19.2.4.2.6.3 Reactor Pedestal/Vessel Supports—Structural Performance Given Failure of BiMAC and Continued Core-Concrete Interactions

In DCD Tier 2, Revision 1, Section 19.3.4, GEH stated that failure of the reactor pedestal, along with the steel liner on it, would constitute a violation of the containment boundary. In NEDC-33201P, Section 21.4, GEH discussed the potential damage from EVE loads on the reactor pedestal and the BiMAC device. This document also discusses an analysis of an LS-DYNA3D model of the pedestal.

The effect of CCI is minimized by the use of a robust reactor pedestal and a refractory material in the floor of the LDW. Since the BiMAC cavity space has a volume of about 400 percent of the full-core melt debris, the possibility for the melt to remain in contact with the reactor pedestal is eliminated. The refractory mat covering the BiMAC pipes is 20-centimeters thick and made of ceramic zirconia with a high melting point temperature, thus eliminating thermal ablation of the basemat by superheated metallic jets resulting from EVE. The ceramic refractory material of the protective layer is susceptible to ablation by superheated oxidic melt impingement. GEH defined the failure threshold for the refractory layer as a remaining thickness of 5 centimeters after ablation and concluded that it would require a melt volume of about 500 tons (full-core debris has 220 tons) to ablate. Analysis showed that the reactor pedestal can withstand ablation of the ceramic refractory material from 20 centimeters down to 5 centimeters. In addition, the analysis showed that the reactor pedestal, which is 2.5-meters thick, can withstand up to a 600 kPa-s impulse. The use of the GDCS deluge and successful BiMAC function quench the corium and reduce the drywell temperature, preventing significant CCI from occurring.

According to the summary of results of severe accident sequence analysis presented in Table 8.3-3 of the PRA report, Revision 1, the concrete ablation caused by CCI 24 hours after onset of core damage is 0.1 meter or less.

19.2.4.3 Staff Evaluation

The structural performance of the containment under severe accident loads reviewed by the staff encompasses (1) the GEH assessment of the Level C (or factored load) pressure capability of the containment in accordance with 10 CFR 50.44(c)(5), (2) the GEH demonstration of the containment capability to withstand the pressure and temperature loads induced by the more likely severe accident scenarios as stipulated in SECY-93-087, Section I.J, (3) the GEH containment structural fragility assessment for overpressurization, and (4) the GEH seismic HCLPF assessment of the RCCV in meeting the SECY-93-087, Section II.N, expectation. The staff also reviewed the GEH assessment of the structural effects of postulated containment phenomenological challenges such as DCH and EVE loads on the containment. The review and evaluation performed in this section were focused on the structural performance of the containment boundary as the ultimate barrier to radionuclide releases to the environment in a severe accident.

The staff reviewed relevant sections of DCD Tier 2 and the PRA report to determine the adequacy and accuracy of the information provided with respect to the performance of various structural components of the containment pressure boundary under severe accident loads. The structural components of the containment that the staff evaluated included the drywell head, RCCV, and reactor cavity structures. The staff evaluation provided in the ensuing sections is based on (1) DCD Tier 2, Revision 4, and Revision 2 of the PRA report, including the information in DCD Tier 2, Sections 6 and 19, and relevant sections of the PRA report regarding the structural containment performance against severe accidents, and (2) the GEH responses to the staff's RAIs.

19.2.4.3.1 10 CFR 50.44 Requirement

GEH addressed the requirements of 10 CFR 50.44 as they relate to hydrogen combustion in DCD Tier 2, Sections 6.2.5.4 and 6.2.5.5. Since the ESBWR containment is inerted, the staff finds that hydrogen burning in the containment is precluded. Further, a necessary condition to deinert the containment is that the containment oxygen concentration increases to 5 percent of the containment volume. DCD Section 6.2.5.5 describes the GEH analysis that determined the time required for the oxygen concentration to increase to the deinerting value of 5 percent. It is significantly greater than 24 hours for a wide range of events, including 100-percent fuel clad-coolant interaction. The staff finds the applicant's analysis to be appropriate and acceptable.

Although the ESBWR containment is inerted and is designed for a DBA pressure of 0.31 MPa gauge (P_d), GEH estimated the containment pressure load resulting from the 100-percent fuel clad-coolant reaction (MWR pressure) to be 0.987 MPa gauge, well above the design pressure.

Based on questions raised during the staff evaluation, GEH resubmitted a revised Level C containment pressure analysis, which is documented in Appendix 19B to DCD Tier 2, Revision 4. The applicant revised other sections of DCD Tier 2, Revision 4, that are related to Level C containment pressure capacity by reference to Appendix 19B.

The GEH Level C containment performance analysis was based on a new and more technically enhanced three-dimensional ABAQUS/ANACAP-U finite element analysis. The staff considers the approach acceptable because the model (1) accounted for the structural characteristics unique to the ESBWR containment (many geometric discontinuities, nonsymmetric loads caused by GDCS pools and pool structures above the top slab, which an axisymmetric finite element model may not be able to capture), (2) properly considered material properties of the structural components, especially with respect to the high-temperature effect, (3) included sufficient mesh refinement to address local stress/strain concentrations, and (4) addressed uncertainty in both finite element modeling and modeling of material properties by using typical industry practice through a lognormal distribution model.

During its review, the staff identified an issue with the new ABAQUS/ANACAP-U analysis concerning the temperature boundary condition of 43.3 °C (110 °F) specified for the drywell head while the rest of the UDW airspace is kept at 260 °C (500 °F) in steady state. Since the drywell head airspace is only separated from the drywell airspace by the bellow which is made of a steel plate, the staff questioned whether the head shell can be kept at 43.3 °C (110 °F) while the drywell airspace is assumed to be at 260 °C (500 °F) steady state. Considering that the refueling pool is located directly above the drywell head, which is kept from being submerged during a postulated beyond-design-basis accident, the drywell head shell is kept from being overheated. The staff believes that the temperature for the drywell head should be determined through an appropriate heat transfer analysis. In RAI 19.2-41, Supplement 2, the staff requested GEH to address this issue. **RAI 19.2-41, Supplement 2 is being tracked as an open item.**

The new ABAQUS/ANACAP-U analysis result shows that, at an internal pressure of 0.987 MPa gauge (MWR pressure), or 3.18P_d, the strain in the liner of the UDW wall at the connection with the top slab reached 0.72 percent, which exceeded the factored load limit for liners (0.3 percent tensile membrane strain, ASME Code, Section III, Division 2, Subarticle CC-3720). The staff questioned the GEH justification for using the thermal-induced strain to reduce the liner strain within the factored load limit. RG 1.7 clearly states that the analysis be performed considering pressure plus dead load alone. Based on the information provided in Figure 19B-5 of DCD Tier 2, Revision 4, the excess liner strain appears to be a localized phenomenon (designated as location A in the figure). It is unclear from the text whether the applicant calculated the strain from the membrane or from the membrane plus bending. In RAI 19.2-86, the staff requested GEH to clarify how it calculated the strain. **RAI 19.2-86 is being tacked as open item.**

Since a correlation has never been established between a particular pressure fragility value and Level C capacity, the staff disagrees with the GEH characterization that the 99-percent confidence fragility value can be used to estimate the actual Level C capacity. Based on Figure 19B-5 of DCD Tier 2, Revision 4, the staff estimates the containment level capacity to be slightly less than 2.5 P_d with respect to the ASME Code allowable, controlled by the failure mode of the liner strain at Location A.

Based on the above discussion, the staff concludes that the GEH approach is acceptable for addressing the 10 CFR 50.44 regulation. However, the calculated containment pressure capacity falls short of the Level C capacity outlined in RG 1.7. Due to the open item that remains to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.2.4.3.2 SECY-93-087 Deterministic Containment Performance Expectation

The staff reviewed the GEH approach to addressing the SECY-93-087 expectation for containment performance (i.e., by referencing an RG 1.7, Revision 3, containment Level C pressure capacity analysis). During its review, the staff identified several issues that should be considered to address SECY-93-087, including (1) identification of the more likely severe accident sequences per SECY-93-087, (2) determination of the containment challenges resulting from the more likely severe accident sequences defined in terms of the transient pressure and temperature time histories (for both short term (up to 24 hours) and long term (up to 72 hours)), and (3) assessment of the containment performance to ensure an adequate margin of the containment Service Level C/factored loads pressure capacity against the severe accident challenges. The Level C containment pressure capability calculation should include the effect of elevated temperature on material properties.

GEH reviewed the accident sequences from the Level 1 PRA and identified the top 10 sequences contributing to CDF. GEH determined that the most likely (97 percent of the core damage sequences identified in the PRA) containment pressure and temperature time histories load resulted from the sequence T_nDP_nIN_TSL (transient with no injection and no depressurization with release category of TSL). The staff considers the selection of the sequence T_nDP_nIN_TSL acceptable, based on enveloping the significant accident sequences as defined in RG 1.200, Revision 1. The initiating event for this sequence is a loss-of-offsite power. The sequence is in Class III. The containment pressure load at 24 hours after onset of core damage is 0.62 MPa and the long-term pressure is below 0.70 MPa. The steady-state temperature for this sequence is below 450 K (350 °F).

GEH calculated of the Level C pressure capacity of the containment based on an axisymmetric ANSYS model and a set of empirical equations (see Sections 19.2.4.2.3 and 19.2.4 of DCD Tier 2, Rev. 0). The staff identified several issues with the GEH approach and associated analysis model for the Level C pressure capacity determination of the containment, which Section 19.2.4.3.3 of this report discusses in detail.

To address the staff's concerns, GEH recalculated the Level C pressure capacity of the containment using a new analysis based on the three-dimensional ABAQUS/ANACAP-U containment structural model and applicable ASME Code equations. The new ABAQUS model uses the pressure and temperature profiles associated with the more likely severe accident sequences and includes detailed modeling of all structural components comprising the containment pressure boundary. During the staff's February 5-7, 2007 onsite audit, GEH presented the analysis model and results of a preliminary ABAQUS analysis of containment performance, which the staff found to be appropriate, except for the temperature of 43.3 °C (110 °F) specified for the drywell head.

The GEH Level C analysis results (19B of DCD Tier 2, Rev. 4) showed that, at an internal pressure of 0.62 MPa gauge (MWR pressure), or $2.0P_d$, the strains in the liners of the three critical locations (DCD Tier 2, Revision 4, Figure 19B-5) of the RCCV wall were kept well below the factored load limit for liners (0.3 percent tensile membrane strain, ASME Code, Section III, Division 2, Subarticle CC-3720). Furthermore, the GEH analysis indicated in Table 19B-6 of DCD Tier 2, Revision 4, that at $2.0P_d$ the induced stresses in RCCV rebar and concrete are significantly less than the ASME Code allowable. The staff concludes that the GEH deterministic containment performance analysis meets the expectation of SECY-93-087, pending the staff's review and acceptance of resolution to RAI 19.2-41, Supplement 2.

19.2.4.3.3 Probabilistic Containment Performance Assessment

GEH performed the containment performance assessment against overpressurization and developed the containment pressure fragility, which is used in the ESBWR Level 2 accident progression analysis. The fragility was developed based on a lognormal distribution, which the staff finds acceptable for the containment pressure capacity.

The use of a lognormal distribution requires a determination of the median values of failure pressure for various CF modes and consideration of variability of the associated parameters. To this end, either a simplified fragility method or a sampling method such as Monte Carlo can be used to establish the containment fragility. To apply the simplified fragility method, the median failure pressure for various CF modes is calculated first and the variability (in both aleatory and epistemic terms) about the median failure pressure is then estimated. The sampling method is implemented using the following steps:

(1) identify all random variables associated with the estimate of the CF pressure
(2) select the probability distribution for each random variable
(3) perform a sampling analysis to determine the containment pressure fragility

In DCD Tier 2, Revisions 0 and 1, and the PRA report, Revision 1, GEH applied the simplified method to establish the containment pressure fragility. GEH relied on an axisymmetric ANSYS finite element analysis of the RCCV and a set of empirical equations for the drywell head to conclude that the containment pressure capacity is controlled by the failure of the drywell head shell. The staff identified several issues with the ANSYS model, which may not be appropriate for capturing the correct CF mode under internal pressurization. Specifically, the lumping of stiffness of the upper slab and the girders, as well as the structures above the upper slab, precludes the determination of the failure of each individual component. The ANSYS model determined that the RCCV pressure capacity was governed by a shear failure of suppression pool slab near the RCCV wall.

The staff also found that the set of empirical equations GEH employed for estimating the drywell head pressure capacity was questionable given the configuration of the ESBWR drywell head shell. The staff noted that GEH based its equations on past studies by Galletly and Shield and Drucker for torispherical shells, however, the test database used to verify these equations is inappropriate for the ESBWR drywell head, which has a much smaller diameter/thickness of shell ratio than those included in the test database. Therefore, use of the empirical equations significantly underestimated the pressure capacity of the drywell head shell.

To address the staff's concerns about the determination of the containment pressure capacity using the ANSYS analysis and the set of empirical equations, GEH revised the estimate of the containment pressure capacity with a new analysis. The applicant documented the new analysis in Appendices 19B (deterministic) and 19C (probabilistic) to DCD Tier 2, Revision 4. The analysis performed was based on a new three-dimensional ABAQUS/ANACAP-U containment structural model. The new ABAQUS model used the pressure and temperature profiles associated with the more likely severe accident sequences and included detailed modeling of all structural components comprising the containment pressure boundary. During the staff's February 5-7, 2007, onsite audit, GEH presented the results of a preliminary ABAQUS analysis of containment performance, which the staff found to be appropriate, except for the temperature of 43.3 °C (110 °F) specified for the drywell head. This issue is discussed in Section 19.2.4.3.1 above. The analysis identified several failure modes that likely control the containment pressure capacity. They are the tensile yielding failure of bolts for the bolted flange system for the drywell head and the shear failure of girders spanning the upper slab.

The GEH analysis for establishing the pressure fragility of the containment system consisted of "best estimate" (median) and uncertainty evaluation, based on a lognormal distribution model. The uncertainty evaluation was performed using the median and an estimate of 95th-percentile pressure capacities. The applicant considered three temperature conditions: (1) steady-state normal operating temperature (ambient), (2) steady-state long-term accident temperature (260 °C or 500 °F), and (3) transient thermal conditions for a temperature spike representative of DCH conditions (peak temperature at 538 °C (1000 °F)). Both median and 95 percent confidence values were developed for the elastic and plastic material properties and failure criteria. For the three temperature conditions, the applicant assembled material data using sources from published literature and NUREG reports. The staff determined that the GEH approach to containment fragility analysis represents a state-of-the-art approach, and both the material data collection and the establishment of the failure criteria for the containment system are reasonable.

Based on the above discussions, the staff concludes that the applicant's containment performance analysis is acceptable, pending resolution of RAIs 19.2-41, Supplement 2.

19.2.4.3.4 Drywell Head

The staff noted that in the PRA report Revision 0, the applicant determined the pressure capacity of the drywell head shell using several empirical equations, which were developed from past studies (Shield and Drucker and Galletly). The staff reviewed the test data, which were the basis for the Galletly Equation (B.8-3), against the parameters for the ESBWR drywell head, which has a D/t ratio of 260, r/D ratio of 0.173, L/D ratio of 0.9, and S_y = 288 MPa (D is the diameter of the cylinder, r is the radius of the knuckle, L is the radius of the sphere, and t is the shell thickness). Among these, D/t and r/D ratios have the most influence on the shell pressure capability. The test data that GEH used, taken from Reference 29, have a minimum D/t ratio of 357. For those test data that have the same r/D ratio as the ESBWR, the corresponding D/t ratio was found to be equal to 2325. Since the ESBWR drywell head shell has a D/t ratio that is well below the minimum D/t ratio found in the test data, the staff questioned the applicant's use of the Galletly Equation (B.8-3) and test data to establish a buckling capacity for the drywell head shell.

In addition, given the high r/D ratio and low D/t ratio for the ESBWR drywell head shell, the torus section of the shell should be very stiff for resisting hoop compression, and the head should fail by inelastic tensile strain in the spherical cap area. Based on this, the staff questioned GEH's pressure capability estimate for the drywell head shell based on the two empirical equations.

Information provided in the GEH response to RAI 19.2-40 identified that the applicant had improved the design of the drywell head shell by adding a taper at the connection with the bolted flanges and the design of the head anchorage by increasing its stiffness. However, based on Tables 19.2-40(1) and 19.2-40(3), submitted as part of the GEH response, the staff questioned whether the drywell head pressure capacity should be controlled by failure of the shell or governed by the capacity of the flange and lower flange plate.

As discussed in Section 19.2.4.4.3, and explained in Appendices 19B and 19C to DCD Tier 2, Revision 4, GEH detailed the three-dimensional ABAQUS/ANACAP-U analysis replaced the empirical equations-based analysis. The ABAQUS analysis verified that the drywell head shell asymmetric buckling cannot precede axisymmetric plastic yielding of the shell in the apex area. The applicant computed the Service Level C capacity of the drywell head shell and supporting components in accordance with ASME Code, Section III. Table 19B-9 of DCD Tier 2, Revision 4, provides the results. The drywell head Level C capacity for the steady-state temperature condition of 260 °C (500 °F) is 1.033 MPa gauge ($3.2P_d$, P_d is the design pressure of 0.31 MPa gauge), which is controlled by the capacity of the inside flange plate of head anchor structure.

GEH provided a more realistic estimate of the failure capacity for the drywell head based on the fragility analysis. The failure state for the drywell head was defined in terms of the leakage assumed to occur because of the yielding of the anchor bolts for the bolted flanges. The HCLPF capacity at 260 °C (500 °F) for the drywell head can be estimated using information provided in Table 19C-10. Given the median capacity at 1.587 MPa gauge and the composite uncertainty of 0.1535, the HCLPF (99-percent confidence value) is estimated to be 1.111 MPa gauge ($3.58P_d$). The staff agrees that the yielding of the bolts for the bolted flanges is likely the failure mode, controlling the ultimate pressure capability of the drywell head. The staff also believes that the release from this failure mode will not result in an uncontrolled large release.

The staff concludes that the reevaluation of the pressure capacity of the drywell head is acceptable, pending resolution of RAI 19.2-41, Supplement 2.

19.2.4.3.4.1 External Water Presence and Temperature during Severe Accidents

The outer surface of the drywell head is immersed in a water pool which provides radiation shielding. The staff identified that the water pool is compartmentalized, is independent of the PCCS/IC cooling pools, and is periodically replenished. The water pool above the drywell head is maintained during and after a severe accident. Therefore, the water pool will limit the temperature rise across the thickness of the drywell head shell. The GEH ABAQUS analysis considered the presence of the water above the drywell head by requiring the temperature of the head shell to be the same as the pool water temperature, which is 43.3 °C (110 °F). The staff questioned the use of 43.3 °C (110 °F) as the temperature of the head shell. This issue is discussed in Section 19.2.4.3.1 above.

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.2.4.3.5 Reinforced Concrete Containment

The staff reviewed the GEH analysis for estimating the internal pressure capacity of the RCCV, which is described in DCD Section 6.2.5.4.2 for Level C/factored load limits, and in Appendix B.8 to the PRA report, Revision 1, for estimating the pressure strength fragility.

GEH provided the analysis results for two loading cases in Tables 19B-6 and 19B-7 based on the detailed three-dimensional ABAQUS/ANACAP-U model. These are 0.62 MPa gauge and 0.992 MPa gauge, representing the internal pressure loads induced from the most likely accident scenarios (SECY-93-087) and the 100-percent MWR pressure (10 CFR 50.44). Tables 19B-6 and 19B-7 compare the maximum stresses in critical areas of the RCCV to Level C allowable limits. Figure 19B-5 identifies the critical strain locations for the RCCV liner, where location A near the top of UDW connecting to the top slab is the critical strain location. For the same location, the vertical inner rebar also showed the highest stress level (Table 19B-6), which could achieve a pressure margin of $3.17P_d$. The liner strain at location A, however, at $2.5P_d$, would just cross over the 0.3-percent Level C limit.

The GEH fragility analysis, as provided in Appendix 19C to DCD Tier 2, Revision 4, identified a similar failure mode for the RCCV as the Level C analysis. Table 19C-8 provides a summary of the pressure fragility for the RCCV and liner. At 260 °C (500 °F) steady-state temperature, the median RCCV pressure capacity and lognormal uncertainty for liner tearing are 1.708 MPa gauge ($5.51P_d$) and 0.1512, respectively. Therefore, the HCLPF pressure capacity for the RCCV is calculated to be 1.2 MPa gauge ($3.877P_d$). The staff concludes that the HCLPF pressure capacity is consistent with the Level C analysis, and the higher HCLPF pressure capacity is achieved because of a realistic limit state of liner tearing strain greater than 2 percent, as opposed to the Level C limit of 0.3 percent.

On the basis of the above discussion, the staff concludes that the GEH approach is acceptable, and the analysis results of the Level C/factored loads pressure capacity and the fragility estimate for the RCCV based on the three-dimensional ABAQUS/ANACAP-U model are acceptable, pending resolution of RAI 19.2-41, Supplement 2

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.2.4.3.5.1 Severe Accident Temperature Loads

The staff reviewed the applicant's analysis of the severe accident temperature loads in the containment and determined that the accident temperature should be clearly defined for evaluating the containment pressure capacity and assessing the potential for thermal-induced containment liner failure.

In Appendix B.8 to the PRA report, Revision 1, GEH characterized the typical temperature of 500 °F for the most likely severe accident scenarios for the ESBWR. The staff requested that GEH clarify the use of the 500 °F value for the most likely severe accident scenarios for the ESBWR. In response to this request, GEH stated that it had reviewed the accident sequences from the Level 1 PRA. GEH identified the top 10 sequences contributing to CDF and determined that the most likely (97 percent of the core damage sequences identified in the PRA) containment pressure and temperature time histories resulted from the sequence T_nDP_nIN_TSL (transient with no injection and no depressurization with release category of TSL). For this sequence, the containment pressure load at 24 hours after onset of core damage is 0.62 MPa and the long-term pressure is below 0.70 MPa; the steady-state temperature is below 450 K (350 °F).

With respect to DCH events, GEH clarified during the February 5–7, 2007, onsite audit that such events are not included in the more likely containment severe accident scenarios, and their occurrence is remote and speculative. DCH is a postulated containment phenomenology event that assumes RPV failure at high pressure (greater than 1 MPa). It constitutes 1 percent of the core damage sequences. The enveloping containment pressure and temperature time histories for 1 percent of the core damage sequences resulted from the sequence T_nDP_nIN_TSL, where the RPV remains at high pressure until lower head failure. For this sequence, the containment pressure load at 24 hours after onset of core damage is 0.72 MPa, and the steady-state drywell temperature is below 500 K (440 °F). The staff finds that the pressure and temperature time-history loads used as input to the structural analysis are appropriate because the estimates of the pressure and temperature during and shortly after the vessel failure, and over the next 1 to 3 days, are consistent with the understanding of severe accident phenomenology and plant systems behavior.

Based on the above discussion, the staff concludes that the GEH assessment of severe temperature loads used as inputs to its containment performance analysis is acceptable, pending resolution of RAI 19.2-41, Supplement 2.

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.2.4.3.5.2 Environment Loads—Seismic: Estimates of Containment Seismic Fragility

The staff reviewed and evaluated DCD Tier 2, Revisions 0 and 1, and Section 15 of the PRA report, Revision 1, with respect to the GEH SMA for Category I structures, including the reinforced concrete containment. The applicant applied the Zion method, described in NUREG/CR-2300, "A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants;" to the seismic HCLPF calculations, and initially used the median ground response spectrum given in NUREG/CR-0098 at 0.3g PGA as the seismic demand. Subsequently, as part of a GEH supplemental RAI response and during discussions with the staff at the February 5-7, 2007, onsite audit, GEH presented a revised SMA. The revision to the

SMA was necessitated by a modification GEH had made to the seismic design ground response spectrum. The new ESBWR design spectrum is specified as the envelope of RG 1.60, anchored at 0.3g PGA, and the North Anna site specific spectrum, anchored at 0.5g PGA. To demonstrate the seismic margin of the ESBWR design, GEH presented seismic HCLPF calculations using the probabilistic variable separation approach for the critical plant SSCs for two separate seismic demand spectra; one for rock sites and one for soil sites. For rock sites, GEH used the North Anna ESP site specific spectrum with a PGA of 0.5g. For soil sites, GEH used a spectrum anchored at 0.3g PGA that envelopes the latest seismic demand spectra for all of the soil sites included in the 28 Central and Eastern United States (CEUS) sites for which EPRI has performed seismic hazard evaluation.

As a part of the SECY-93-087 expectation for seismic margin assessment of evolutionary and ALWR designs, the plant level HCLPF value should be demonstrated up to approximately 1.67 times the design-basis SSE. The ESBWR design-basis SSE is defined by the response spectra shown in DCD Tier 1, Figures 5.1-1 and 5.1-2. The staff noted that demonstration of seismic margin using two separate response spectra does not appear to satisfy the expectation of SECY-93-087. The staff believes that two issues in the seismic margin assessment need to be addressed in this regard—(1) the shape of the review-level earthquake spectrum should be compatible with the design-basis spectrum, and (2) the review-level earthquake PGA should be selected to be about 1.67 times the design-basis PGA.

To address the issues raised by the staff, in DCD Tier 2, Revision 4, Section 19, and Section 15 of the PRA report, Revision 2, GEH developed a performance-based seismic response spectrum (PBRS), which is the same as the certified seismic design response spectrum (CSDRS) at and above 9 Hz. For frequencies below 9 Hz, the PBRS bounds all soil sites in the CEUS except Vogtle; however, it falls slightly below the CSDRS for frequencies below 9 Hz. Even though GEH made significant improvement, it still falls short of meeting the guidance of SECY-93-087 which, in the certified design, expects the CSDRS to be used for the margin assessment. The staff also requested that Chapter 15 of NEDO-33201 Revision which the SMA assessment should be properly referenced in DCD Tier 2, Revision 4, Section 19. The staff considers this issue to be unresolved at this time. **RAI 19.2-90 and 19.2-92 are being tracked as open items.**

The staff also found an inadequate modeling of the fault tree for the fire protection water system (FPWS), which was utilized in the seismic event trees for both full power and shutdown operations. Figure 15-15 of NEDO-33201 Revision 2 shows that the FPWS fault tree has only one component: pump. The staff noticed that FPWS is located in the Fire Water Service Complex (FWSC), which is designed and analyzed using the CSRDS. FWSC consists of two waster storage tanks, a pump enclosure and attached piping. To ensure successful vessel water injection, all three components must remain functional during and after a seismic event. Therefore, the fault tree for FPWS should have all three components in OR gates. The staff requested the applicant to make correction to the FPWS fault tree and to provide a revised HCLPF calculation. The staff considers this issue to be unresolved at this time. **RAI 19.2-91 is being tracked as an open item.**

The staff finds that the seismic margin assessment based on PRA seismic sequences, as described in Section 15 of the PRA report, Revision 2, to be in accord with SECY-93-087. However, the applicant only estimated HCLPFs for five structural components by analysis, while assuming that the remaining SSCs had HCLPFs equal to 0.84g PGA. Since the DCD provides no enforceable procedure to ensure that all SSCs identified on the seismic sequences will be qualified for HCLPF capacity equal to or greater than 1.67 times the ESBWR CSDRS, the staff believes that the COL applicants or holders will have an alternative to qualify SSC HCLPFs with respect to the site specific GMRS, which is generally less than CSDRS. This issue is further discussed in RAI 19.2-90. Until this issue is resolved, the staff cannot conclude that the ESBWR DC has a seismic margin equal to 1.67 times the CSDRS.

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.2.4.3.5.3 Containment Liner

The staff reviewed the applicant's analysis of the containment liner integrity to maintain a leaktight condition under severe accident loads. The containment liner is anchored to the reinforced concrete wall by regularly spaced T-bar stiffeners with webs welded to the liner. The T-bar stiffeners are embedded in the concrete. The T-bar stiffeners are spaced 50 centimeters apart. Thermal and pressured induced liner failure should be assessed. The greatest concern is at major penetrations, where stress concentrations and constraint to thermal growth are expected.

The staff questioned the applicant's use of a 21-percent ultimate fracture strain criterion for the liner material in the fragility analysis. The 21-percent strain for the liner material (SA-516 Gr. 70) is based on the material specification in ASME Code, Section III, Part A. The staff noted that 21 percent is the minimum required elongation in a 2-inch uniaxial test coupon, and 17 percent is the minimum required elongation in an 8-inch uniaxial test coupon. The liner is subjected to biaxial state of stress and strain concentrations near major penetrations. The staff concluded that the maximum liner strain should not exceed 10 percent.

To assess the liner failures induced by high temperature loads in a DCH event, the applicant estimated the liner failure strain at 1100 °F to be about 23 percent, based on the available test data for SA533 and A36 steel. The staff found that these tests were performed using specimens typically 2 inches or less, and they do not consider a biaxial state of stress. The staff concluded that the maximum liner strain should not exceed 11 percent (onset of void nucleation) at the DCH temperature.

In discussions with the staff during the February 5-7, 2007, onsite audit, GEH agreed to use the factored load limits of ASME Code, Section III, Division 2, for the deterministic assessment of liner integrity, and to use an 8-percent failure strain limit for the liner plate in the three-dimensional ABAQUS/ANACAP-U fragility analysis. The staff noted that in Appendices 19B and 19C to DCD Tier 2, Revision 4, GEH used ASME Code, Section III, Division 2, Subarticle CC-3720, limits for liners in the deterministic analysis and much less than 8 percent strain (Table 19C-5) in the fragility analysis. The staff determined that the applicant's approach is acceptable and resolves the two items discussed above.

19.2.4.3.5.4 Penetrations

The staff reviewed and evaluated the applicant's evaluation of the leakage potential of operable penetrations induced by the accident pressure and temperature. In Section B.8.2.2.2 of the PRA report, Revision 1, GEH used a SANDIA-proposed springback methodology to assess leakage prevention at seals. According to Section 8.2.1.3 of the PRA report, Revision 0, the allowable TSL is 0.5 percent of containment air volume per day at rated pressure, and based on MAAP test runs, the effective flow area required to allow 0.5 percent of the containment air volume to leak per day at design pressure is approximately 3.4×10^{-6} m^2 (3.4 mm^2). In

Section B.8.2.2.2 of the PRA report, Revision 1, GEH estimated the leakage potential for the drywell head with a 10.4-meter diameter, two drywell equipment hatches with a 2.4-meter diameter and one wetwell hatch with a 2.0-meter diameter. According to the GEH calculation presented in Section B.8.2.2.2 of the PRA report, Revision 1, the separation displacement at 1.204 MPa capability pressure is calculated to be about 0.146 millimeters for the drywell head and 0.204 millimeters for the most flexible hatch. A comparison of the separation displacements of the hatches with the springback limit (0.127 millimeters) for leakage initiation, shows that the leakage gap for the drywell head is 0.019 millimeters and the leakage gap for drywell hatches is 0.077 millimeters. Although the leakage gap of 0.019 millimeters appears to be small for the drywell head, the leakage area using a 10.4-meter diameter for the drywell head is estimated to be 465 mm^2, which is much larger than the 3.4 mm^2 allowed for TSL.

In discussions with the staff during the February 5-7, 2007, onsite audit, GEH stated that the three-dimensional ABAQUS/ANACAP-U analysis will be used to provide the assessment of leakage potential of major penetrations. GEH also stated that bolts for equipment hatches and the drywell head will be preloaded to ensure no leakage gaps at the Level C pressure. GEH also presented the preliminary assessment results of major penetrations, which the staff found acceptable. GEH provided the new leakage assessment of equipment hatches and the drywell head in Appendices 19B and 19C to DCD Tier 2, Revision 4, for deterministic and probabilistic analyses, respectively.

The applicant based its deterministic analysis on the ASME Code, Section III, requirements for Level C capacity determination. For the fragility analysis, GEH constructed detailed local finite element models and applied the response from the global ABAQUS model to the local models as boundary conditions. The hatch in the UDW was chosen as the basis of the modeling, since all equipment hatches have similar configurations. Furthermore, the equipment hatch in the LDW differs only in that it penetrates the thicker pedestal wall while the thinner RCCV wall in the UDW is more flexible and more critical for deformation leading to possible flange distortions or tearing in steel components of the hatch. In addition, the LDW hatch has a closure lid on the inside of the containment so that the internal pressure keeps the inner seal closed and prevents the interior of the penetration from being exposed to high temperatures. Table 19C-5 provides the failure criteria (or limit states) for leakage from either tearing of steel components or flange distortion and loss of seal. The tearing is in terms of strains, while the flange separation is indicated by the first yield in bolts. The staff finds the criteria in Table 19C-5 acceptable. The fragility analysis results provided in Table 19C-11 indicate that the pressure capacity of equipment hatches is controlled by leakage from flange distortion with a median value of 1.882 MPa gauge (6.07 P_d) at 260 °C (500 °F) and a composite uncertainty of 0.1542. Therefore, the HCLPF pressure capacity for equipment hatches can be inferred to be 1.315 MPa gauge (4.24 P_d).

The staff concludes that the assessment of leakage potential of major penetrations using the three-dimensional ABAQUS/ANACAP-U model is appropriate and acceptable. In addition, the equipment hatches appear to be rugged in resisting the internal pressure up to 4.24 P_d at an accident temperature of 260 °C (500 °F). However, the staff's conclusions are pending resolution of RAI 19.2-41, Supplement 2, discussed in section 19.2.4.3.1 above.

19.2.4.3.6 Reactor Cavity Structures

The staff reviewed the applicant's analysis of the potential failure of the reactor cavity structures subjected to postulated EVE loadings. EVE is a postulated internal initiated event of energetic FCI. It is triggered by the melt released from the failed RPV lower head falling into and traversing the depth of a preexisting water pool in the LDW cavity. The result of EVE events is energetic pressure pulses, with magnitudes in the kilo-bar range, which are potentially capable of loading major structures to failure when large quantities of melt react with highly subcooled water. The EVE loading is characterized by the impulse (the time-integral of the pressure) acting on the surface of a structure.

The BiMAC device consists of a layer of thick-walled steel pipes embedded in reinforced concrete that supports them in all directions. BiMAC is also protected by a 0.2-meter-thick sacrificial refractory layer of ceramic zirconia material. The RPV support brackets are made of structural steel and provide structural support to the RPV and the reactor shield wall.

19.2.4.3.6.1 Reactor Cavity—Structural Performance under Ex-Vessel Steam Explosion Loadings

The ESBWR LDW is designed with a large cavity space. The key parameter for EVE is the depth of the preexisting subcooled water pool in the LDW cavity. In the GEH PRA analysis for severe accident sequences, the reactor cavity structures and penetrations are considered to be failed when the water depth is greater than or equal to 1.5 meters. GEH estimated that, for those sequences in which the water level is greater than 1.5 meters, the contribution to core damage from sequences to be considered for the EVE constitutes only 0.9 percent of CDF. The GEH assumption of CF from EVE sequences with a water level at 1.5 meters is conservative, since the closest equipment hatch in the LDW cavity is located 2.2 meters above the 1.5-meter critical depth of water for EVE assessment. The equipment hatch will not likely be impacted by the EVE for the subcooled water pool with a depth less than 1.5 meters; however, the equipment hatch is the likely CF path for a water depth greater than 1.5 meters. The staff noted that the design of the 2.4-meter-thick reactor pedestal is robust. Also, the large space of the BiMAC cavity (90 cubic meters), which is sufficient to accommodate about 400 percent of the full-core debris, is protected by a layer (0.2-meter thickness) of sacrificial refractory material (ceramic zirconia), that should protect the basemat and lower pedestal from the CCI effect.

For a water depth less than 1.5 meters, the sequences involved constitute 99 percent of CDF. GEH performed a PM-ALPHA.L-3D analysis to characterize the EVE pressure loads on the side and base of the cavity, and performed a DYNA-3D analysis to quantify the structural capacity of the pedestal and BiMAC against EVE pressure impulse. In Section 21.4.4.5 of the PRA report, Revision 1, GEH estimated that the reactor pedestal pressure capacity has a margin of 5 times against EVE pressure loads. GEH concluded that the failure of cavity structures from EVE events is PU. The staff finds that the pedestal and other cavity structures have a sufficient structural capacity to resist EVE pressure load and concurs with the applicant's conclusion regarding EVE-induced failure of cavity structures.

On the basis of the above discussion, the staff concludes that the GEH evaluation of the reactor cavity structures is acceptable.

19.2.4.3.6.2 BiMAC Device—Structural Performance under Ex-Vessel Steam Explosion Loadings

The staff reviewed the applicant's assessment of the structural integrity of the BiMAC device under EVE loading. The BiMAC is unique to the ESBWR design. The BiMAC is a passive safety system which cools the core melt ejected from an RPV lower head failure and reduces the impact of CCI. Upon detecting the melt deposited on the reactor cavity floor, the thermal couples embedded in the cavity floor send a signal to open the squib-activated valves for the

GDCS water supply to the BiMAC. The objective is to prevent the possibility of releasing water to the cavity floor before core melt ejection. The BiMAC is protected by a 20-centimeter-thick sacrificial layer of ceramic zirconia with a melting temperature range of 2950–3120 K. This extremely high melting temperature minimizes the possibility of thermal ablation. GEH also analyzed the effect of oxidic ablation of the sacrificial protective layer. GEH used 5 centimeters of remaining thickness as the structural failure threshold for the sacrificial layer. The applicant's analysis concluded that it would require a melt of 500 tons (more than twice the full-core debris) to penetrate the 20-centimeter sacrificial layer down to within 5 centimeters of the BiMAC pipes. The staff finds acceptable the GEH conclusion that BiMAC failure by melt impingement is PU.

GEH also assessed the effect of EVE on the function of the BiMAC and concluded that, for the lower water depths (less than 1.5 meters), the BiMAC structural capacity is more than 8 times the pressure demand induced by the EVE event, and failure is PU. For high water depth (more than 1.5 meters), which constitutes only 0.9 percent of CDF, GEH conservatively assumed that the BiMAC failed. The staff finds the applicant's assessment of BiMAC failure from EVE events to be acceptable.

19.2.4.3.6.3 Reactor Pedestal/Vessel Supports—Structural Performance Given Failure of BiMAC and Continued Core-Concrete Interactions

GEH confirmed that the failure of the BiMAC constitutes a breach of the containment boundary and modeled it in the Level II PRA accident progression analysis. The staff finds the GEH approach to be acceptable.

19.2.4.4 Conclusions

Section 19.2.4 of this report provides the staff's review and assessment of the applicant's evaluation of the ESBWR containment structural performance. The staff focused its review on the ability of the structural components comprising the containment pressure boundary to meet the (1) 10 CFR 50.44 requirement, (2) SECY-93-087 expectation for deterministic containment performance, and (3) SECY-93-087 expectation for seismic margin assessment. The staff's review also focused on assessing the adequacy of the applicant's evaluation of containment pressure fragility.

The staff reviewed the applicant's approach, as documented in Appendices 19B and 19C to DCD Tier 2, Revision 4, for the deterministic and probabilistic containment pressure capability assessments and finds it acceptable. GEH utilized the state-of-the-art finite element method as implemented in ABAQUS/ANACAP-U to address the complex nonlinear structural behavior of the reinforced concrete containment when pressurized several times above the design limit. However, the staff identified several open issues associated with assumptions and analysis results of the applicant's containment ultimate pressure capacity assessment which require further evaluation. These issues are listed as open items in the associated sections.

On the basis of its review and assessment, the staff concludes that the acceptance of the applicant's containment performance evaluation to meet the requirement of 10 CFR 50.44, the SECY-93-087 expectation for containment structural performance, and the staff's expectation of the quality of the containment fragility analysis will be contingent upon the successful resolution of the open issues identified in the sections above.

19.2.5 Accident Management

Accident management consists of the actions taken by the plant's emergency response organization (including plant operations, technical support, and management staff), to prevent core damage, terminate core damage once it begins, maintain containment integrity, and minimize offsite radiation releases. Severe accident management refers to those actions that would mitigate the consequences of accidents that result in core damage. The objectives of a severe accident management program are to arrest core melt progression by cooling the molten core material, either in-vessel if possible, or ex-vessel if the debris has entered the containment building, and to ensure that fission products are not released to the environment. The ultimate objective is to achieve a safe, stable state. To accomplish these objectives, the emergency response organization should make full use of the plant's design features, including both standard and non-standard use of plant systems and equipment.

The nuclear power industry initiated a coordinated program on accident management in 1990 (Section 5 of NEI 91-04, Revision 1, "Severe Accident Closure Guidelines," lays out the elements of the industry's severe accident management closure actions that have been accepted by the NRC). This program involves the development of (1) a structured method by which utilities may systematically evaluate and enhance their abilities to deal with potential severe accidents, (2) vendor-specific accident management guidelines for use by individual utilities in establishing plant-specific accident management procedures and guidance, and (3) guidance and material to support utility activities related to training in severe accidents. Using the guidance developed through this program, each operating plant has implemented a plant-specific accident management plan as part of an industry initiative.

Based on its reviews of these efforts, severe accident evaluations in Individual Plant Examinations (IPEs), and industry PRAs, the NRC staff has concluded that improvements to utility accident management capabilities could further reduce the risk associated with severe accidents. Although future reactor designs such as the ESBWR will have enhanced capabilities for the prevention and mitigation of severe accidents, accident management will remain an important element of defense-in-depth for these designs. However, the increased attention on accident prevention and mitigation in these designs can be expected to alter the scope and focus of accident management relative to that for operating reactors. For example, increased attention on accident prevention and the development of error-tolerant designs can be expected to decrease the need for operator intervention, while increasing the time available for such action if necessary. This will tend to make it less likely for the emergency response organization to make rapid decisions and permit a greater reliance on support from outside sources. For longer times after an accident (several hours to several days), the need for human intervention and accident management will continue.

For both operating and advanced reactors, the overall responsibility for accident management, including development, implementation, and maintenance of the accident management plan, lies with the nuclear utility, because the utility bears ultimate responsibility for the safety of the plant and for establishing and maintaining an emergency response organization capable of effectively responding to potential accident situations. For operating plants, vendors have played key roles in providing essential severe accident management guidance and strategies for implementation. This guidance has served as the basis for severe accident management procedures and for training utility personnel in carrying out the procedures. Computational aids for technical support have been developed, information needed to respond to a spectrum of severe accidents has been provided, decision-making responsibilities have been delineated, and utility self-evaluation methodologies have been developed and utilized.

The staff requested additional information on the process that will be used by GEH to develop the Severe Accident Guidelines (SAGs) in RAI 19.2.4-1 and its supplements. In response to RAI 19.4.2-1, Supplement 2, GEH provided additional details regarding development of the SAGs and referred to the ESBWR Human Factors Engineering Procedures Development Implementation Plan, NEDO-33274, Revision 2, March 2007, that presents the procedure development processes and methodologies to be used in the development of procedures including ESBWR SAGs and the ESBWR Severe Accident Management Guidelines (SAMGs) derived from them.

The staff is currently reviewing GEH's response. **RAI 19.4.2-1 is being tracked as an open item**.

Due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.2.6 Consideration of Potential Design Improvements under 10 CFR 50.34(f)

19.2.6.1 Regulatory Criteria

In 10 CFR 50.34(f)(1)(i), the NRC requires an applicant to "perform a plant/site specific PRA, the aim of which is to seek such improvements in the reliability of core and CHR systems as are significant and practical and do not impact excessively on the plant." The applicant provided an initial evaluation of potential design improvements (severe accident mitigation design alternatives (SAMDAs)) for the ESBWR in response to RAI 19.4-1.

To address questions raised by the staff on the initial evaluation, GEH provided a revised RAI response dated August 14, 2007. In this response, the applicant concluded that, because of the small risk associated with the ESBWR design, a majority of the design improvements beyond those that already exist as part of the design were either of a procedural and administrative nature or were not considered to be cost beneficial. The review of the applicant's evaluation is presented below.

19.2.6.2 Summary of Technical Information

19.2.6.2.1 Estimate of Risk for the ESBWR

As stated earlier in Section 19.1.4.3, the applicant provided an estimate of the offsite risk to the population within 80 kilometers (50 miles) of the site in Section 10 of DCD Tier 2. Table 19.1-10 summarizes the baseline results for internal events occurring during full-power operation and compares them to the NRC's individual and societal safety goals. The results indicate that the risk from severe accidents would be at least 4 orders of magnitude lower than the NRC's safety goals.

For external events and shutdown modes, the PRA includes surrogate values for all but seismic events. Section 19.1.4.5 lists the external event and shutdown CDF and LRF results. The values listed show the same magnitude as those for the at-power internal events case. Because the individual CDF values are developed with differing levels of conservatism, the applicant indicated that it is not meaningful to add the CDF or LRF values to create total values. Nevertheless, it is apparent that for the two safety goal measures the total risk from all accidents (internal and external events) would not increase by more than 2 orders of magnitude.

GEH affirms that the individual risk and societal risk goals are maintained with sufficient margin.

The risk results, together with supporting sensitivity studies, lead to the risk insight that the public health and safety is well achieved in the ESBWR design, as shown by the PRA analysis.

19.2.6.2.2 Identification of Potential Design Improvements

The applicant identified 177 candidate design alternatives based on a review of design alternatives for other plant designs, including the license renewal environmental reports and the GEH ABWR SAMDA study. The applicant eliminated certain design improvements from further consideration on the basis that the ESBWR design already incorporates them. The following are examples of design enhancement features currently included in the design:

- improved IC design
- DPVs
- ac-independent fire water pumps for makeup and injection
- passive containment cooling system
- BiMAC device and GDCS deluge function
- dc power reliability
- actuation logic reliability
- motor-driven feedwater pumps
- water pool above drywell head
- containment ultimate strength and maximum design pressure
- incorporation of flood mitigation into design
- RWCU heat exchanger sized for DHR
- 72-hour coping period for SBO
- upgraded low-pressure piping for the RCPB
- digital I&C

The applicant's screening process eliminated 42 potential alternatives as being inapplicable, 65 design alternatives were considered to be similar to those already included in the ESBWR design, 29 items were marked as procedural or administrative as opposed to design features (whose benefits were considered to be unlikely to exceed those alternatives evaluated relative to their potentially high costs), and 26 items were ruled out on the basis of their high cost relative to potential benefits. The remaining 15 issues were considered to have very low benefit because of their insignificant contribution to reducing risk.

19.2.6.2.3 Risk Reduction Potential of Design Improvements

The applicant assumed that each design alternative would work perfectly to completely eliminate all severe accident risk from evaluated internal events. This assumption is conservative as it maximizes the benefit of each design alternative. The applicant estimated the public exposure design alternative benefits on the basis of the reduction of risk expressed in terms of whole body person-rem per year received by the total population within an 80-kilometer (50-mile) radius of the ESBWR plant site, as discussed in Section 19.2.6.2.1 of this report.

The applicant used the cost-benefit methodology found in NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," issued in 1997 (Ref. 19-19), to calculate the maximum attainable benefit associated with completely eliminating all risk for the ESBWR. This methodology considers averted onsite and replacement power costs. The applicant estimated the present worth of eliminating all severe accident risk to be about $4630. If the offsite population doses and property damage costs were to be a factor of 10 higher, the applicant estimated a present worth of about $41,380 (see Section 19.2.6.5).

The applicant's risk reduction estimates are based on point-estimate (mean) values, without consideration of uncertainties in CDF or offsite consequences. Even though this approach is consistent with that used in previous design alternative evaluations, further consideration of these factors could lead to significantly higher risk reduction values, given the extremely small CDF and risk estimates in the baseline PRA. In assessing the risk reduction potential of design improvements for the ESBWR, the NRC staff has based its evaluation on the applicant's risk reduction estimates for the various design alternatives, in conjunction with an assessment of the potential impact of uncertainties on the results. Section 19.2.6.7 discusses further this assessment.

19.2.6.2.4 Cost Impacts of Candidate Design Improvements

NEDO-33306, Revision 1, Licensing Topical Report, "ESBWR Severe Accident Mitigation Design Alternatives," issued August 2007 did not assess the capital cost associated with the various design alternatives evaluated by the applicant for the ESBWR. Instead, the applicant maintained that the economic impacts of severe accidents, when combined with their associated frequencies, result in an overall risk that is significantly lower than current operating reactors, therefore, making any additional design modifications costly as compared to any potential benefits.

On the basis of the analyses performed by GEH, the NRC staff views the applicant's assertion of potential costs for the ESBWR, as compared to those evaluated for existing plants, as acceptable, given the wide disparity between the cost of design implementation and the potential benefits (i.e., even with large conservatisms in the assessment of benefits, they are too low when compared with typical costs).

19.2.6.2.5 Cost-Benefit Comparison

The methodology used by GEH was based primarily on the NRC's guidance for performing cost-benefit analysis outlined in NUREG/BR-0184. The guidance involves determining the net value for each SAMDA according to the following formula:

Net Value = (APE + AOC + AOE + AOSC) - COE

Where:
APE = present value of averted public exposure ($)
AOC = present value of averted offsite property damage costs ($)
AOE = present value of averted occupational exposure costs ($)
AOSC = present value of averted onsite costs, ($). This includes cleanup and decontamination and long-term replacement power costs.
COE = cost of enhancement ($)

If the net value of a SAMDA is negative, the cost of implementing the SAMDA is larger than the benefit associated with the SAMDA and it is not considered to be cost beneficial. Table 19.2-2 summarizes the applicant's and NRC staff's estimates of each of the associated cost elements. The provided results are based on the approach, parameters, and data listed in NUREG/BR-0184. As indicated above, the GEH estimates are based on the Revision 1 CDF (a value of 7.54×10^{-8}/yr). The results presented for the NRC staff have adjusted the estimated present value using the revised CDF (a value of 1.2×10^{-8}/yr).

The applicant provided estimates using a 7-percent discount rate. The NRC recently revised Revision 4 of NUREG/BR-0058, “Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission,” in August 2004, to reflect the agency's policy on discount rates. NUREG/BR-0058, Revision 4 states that two sets of estimates should be developed—one at 3 percent and one at 7 percent. Use of a 3 percent discount rate would result in an almost doubling of the estimated benefits

Table 19.2-2 Summary of Estimated Averted Costs

Quantitative Attributes		**Present Value Estimate ($)**		
		Best [a]	**Maximum** [b]	**NRC Staff Maximum** [c]
Health	Public	366	3,660 [d]	11,400
	Occupational	38	76	24
Property	Offsite	157	1,570 [d]	7,010 [e]
	Onsite	NA [f]	NA [f]	NA [f]
Cleanup and Decontamination	Onsite	1,167	1,591	510
Replacement Power		2,900	34,486	9,350
Total		**4,628**	**41,383**	**28,290**

[a] “Best estimate” is based on mean release frequency (from Revision 1 of the PRA) and “best estimate” parameter values.

[b] Maximum estimate is based on mean release frequency (from Revision 1 of the PRA) and high estimate parameter values.

[c] Reviewer maximum is based on parameter values used in b, release frequency (from Revision 2 of the PRA), and a 3-percent discount rate.

[d] Estimate is based on a factor of 10 increase in estimated dose or public property.

[e] Estimated using the applicant-provided EPRI ALWR URD, property damage, and the new release category frequencies.

[f] Not Analyzed.

It is important to note that the monetary present value estimate for each risk attribute does not represent the expected reduction in risk resulting from a single accident. Rather, it is the present value of a stream of potential losses extending over the projected lifetime (in this case, 60 years) of the facility. Therefore, it reflects the expected annual loss resulting from a single accident, the possibility that such an accident could occur at any time over the licensed life, and the effect of discounting these potential future losses to present value.

As indicated above, the applicant estimated the total present dollar value equivalent associated with complete elimination of severe accidents at a single ESBWR unit site to range between $4,628 and $41,383. The estimated cost of replacement power has the largest effect on the averted cost. For any SAMDA to be cost beneficial, the enhancement cost must be less than $41,383. Based on this, the applicant concluded that none of the SAMDA candidates are cost beneficial.

19.2.6.3 Staff Evaluation

In 10 CFR 50.34(f)(1)(i), the NRC requires an applicant to perform a plant- or site specific PRA. The aim of this PRA is to seek improvements in the reliability of core and CHR systems that are significant and practical and do not impact excessively on the plant. On the basis of its review, the staff concludes that the ESBWR PRA and the applicant's use of the insights of this study to improve the design of the ESBWR meet this requirement.

The set of potential design improvements considered for the ESBWR includes those from generic BWR SAMA reports and from the ABWR design. The ESBWR design already incorporates several design enhancements relative to severe accident mitigations. These design improvements have resulted in a CDF that is about an order of magnitude less than that of the ABWR design. For example, the ESBWR design can cope with an SBO for 72 hours (i.e., no reliance on ac power for the first 72 hours), eliminating CDF sequences that contributed more than 40 percent of CDF in the ABWR design.

The staff considers the applicant's review of the potential SAMDA and their impacts on the ESBWR design acceptable. The staff's review did not reveal any additional design alternatives that the applicant should have considered.

The applicant's estimates of risk do not account for uncertainties either in CDF or in offsite radiation exposures resulting from a core damage event. The uncertainties in both of these key elements are fairly large because key safety features of the ESBWR design are unique, and their reliability has been evaluated through analysis and testing programs rather than through operating experience. In addition, the estimates of CDF and offsite exposures do not account for the added risk from earthquakes.

The staff's analyses of the total present value using the mean CDF and release frequencies from Revision 2 of the PRA and a 3-percent discount rate indicate a maximum value of about $28,290. If one were to apply an order of magnitude increase because of uncertainties in release and CDFs, the total present worth value would be about $283,000. Additionally, if one were to adjust annual replacement power cost for future energy cost increase, the total present dollar value would be even higher.

GEH indicated that any design modifications, including even a change of a manufacturer to reduce CCF, would incur approximately $2 million associated with the implementation of the supplier quality assurance program. The staff's review concurs with the applicant's conclusion that none of the potential design modifications evaluated could be justified on the basis of cost-benefit considerations. The staff further concluded that it is unlikely that any other design changes would be justified on the basis of person-rem exposure considerations because the estimated CDF would remain very low on an absolute scale.

19.2.7 Conclusions

As discussed in Section 19.1 of this report, the applicant made extensive use of the results of the PRA to arrive at a final ESBWR design. As a result, the estimated CDF and risk calculated for the ESBWR design are very low. The low CDF and risk for the ESBWR design are a reflection of the applicant's efforts to systematically minimize the effect of initiators/sequences that have been important contributors to CDF in previous BWR PRAs. This minimization has been done largely through the incorporation of a number of hardware improvements in the ESBWR design. Section 19.1 of this report discusses these improvements and the additional ESBWR design features that contribute to low CDF and risk for the ESBWR.

Because the ESBWR design already contains numerous plant features oriented toward reducing CDF and risk, the benefits and risk reduction potential of additional plant improvements is significantly reduced. This reduction is true for both internally and externally initiated events. Moreover, with the features already incorporated in the ESBWR design, the ability to estimate CDF and risk approaches the limitations of probabilistic techniques.

However, due to the open items that remain to be resolved for this section, the staff was unable to finalize its conclusions regarding acceptability.

19.3 References

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